LOUISIANA POWER & LIGHT
MIDDLE SOUTH UTILITIES SYSTEM
317 BARONNE STREET • P. O. BOX 60340
NEW ORLEANS, LOUISIANA 70160 • (504) 595-2781

J. G. DEWEASE
SENIOR VICE PRESIDENT-
NUCLEAR OPERATIONS

May 27, 1988

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QA

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: Waterford SES Unit 3
Docket No. 50-382
Technical Specification Change Request NPF-38-83

Gentlemen:

Louisiana Power & Light files this application for an emergency amendment to the Waterford 3 Technical Specifications in accordance with 10 CFR Part 50.91(a)(5). The amendment which does not involve an unreviewed safety question nor a significant hazards consideration would revise the allowable CEA drop time of Tech Spec 3.1.3.4.

In anticipation of the performance of the surveillance for Technical Specification 3.1.3.4 (scheduled for May 27, 1988), some CEAs may exceed the allowable drop time (LP&L will provide the test results upon completion of the surveillance). To resolve these discrepancies between the Tech Spec allowable drop time and the measured drop times, LP&L and Combustion Engineering (CE) have reviewed the affected safety analyses to support increasing the allowable drop time to 3.2 seconds.

CEA drop time testing, by nature of the test, is performed just prior to restart. A normal operating temperature and pressure environment is required by Tech Spec in order to conduct the test. In other words, plant criticality (mode 2) and subsequent power operation is imminent. A failure to meet the LCO will prohibit the plant from entering into Mode 2 and thus prevent power operation.

The entry into Mode 2 is scheduled for May 27 - 30, 1988. The LP&L and Middle South Utilities systems experience the highest demand for electrical power during the upcoming summer months. The unavailability of Waterford 3 would require the operation of more costly fossil fired units or the open market purchase of wholesale electricity at prices much higher than Waterford 3 electricity. The alternative to nuclear generated electricity represents an undue economic burden upon LP&L and its ratepayers.

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In anticipation of a potential unsuccessful CEA drop time test this proposed change is submitted. A similar test result at ANO-2 on or about May 5, 1988, uncovered previously unrecognized conservatisms in the testing method which led to an approved emergency Technical Specification change for ANO-2 on May 12, 1988.

Forewarned by the problems experienced by ANO-2, and having less than one month before similar testing was to be conducted at Waterford 3, LP&L immediately reviewed the previous Waterford 3 test results and noted that several CEA drop times were close to the 3.0 second Technical Specification LCO. Consequently, as a prudent measure, LP&L requested CE to initiate evaluations necessary to determine if a similar LCO change could be justified in the event that the upcoming drop test failed to meet the 3.0 second LCO. The CE/LP&L evaluation, which was completed a matter of hours prior to performing the drop test on May 27, 1988, forms the basis for the enclosed Technical Specification change request.

Should there be any questions, please contact Larry Laughlin at (504) 464-3499.

Very truly yours,

J.G. Dewease

J.G. Dewease
Senior Vice President -
Nuclear Operations

JGD:SEF:ssf

Attachment: Affidavit
NPF-38-83

cc: R.D. Martin, J.A. Calvo, D.L. Wigginton, NRC Resident Inspectors Office, E.L. Blake, W.M. Stevenson

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the matter of	)
	)
Louisiana Power & Light Company	) Docket No. 50-382
Waterford 3 Steam Electric Station	)

AFFIDAVIT

J.G. Dewease, being duly sworn, hereby deposes and says that he is Senior Vice President-Nuclear Operations of Louisiana Power & Light Company; that he is duly authorized to sign and file with the Nuclear Regulatory Commission the attached Technical Specification Change Request NPF-38-83; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information and belief.

J.G. Dewease
J.G. Dewease
Senior Vice President-Nuclear Operations

STATE OF LOUISIANA)
) ss
PARISH OF ORLEANS)

Subscribed and sworn to before me, a Notary Public in and for the Parish and State above named this 14th day of July, 1988.

Nancy Hull Troker
Notary Public

My Commission expires life.

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-38-83

The proposed change would revise the allowable CEA drop time of Technical Specification 3.1.3.4.

Existing Specification

See Attachment A.

Proposed Specification

See Attachment B.

Description

Waterford 3 Technical Specification (TS) 3.1.3.4 presently requires the 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 520°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 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. The proposed change would revise the CEA drop time to 3.2 seconds.

CEA drop times are measured in accordance with TS 4.1.3.4 requirements. The surveillance testing method (which was also utilized during the first refueling outage) uses special software loaded on one of the Control Element Assembly Calculators (CEACs) which turns the selected CEAC into a specialized high speed data acquisition system capable of the simultaneous monitoring of all 91 CEA positions every 50 milliseconds through their individual reed switch position transmitters. The data may then be transferred to a floppy disk for permanent storage or analysis. The special software (CEA Drop Time Test, or CDTT software) initiates the test by transmitting a large penalty factor to each of the Core Protection Calculator (CPC) channels, producing a reactor trip. It should be noted that the point at which power is interrupted to the CEA drive mechanism is the reactor trip breaker, not the individual breakers.

To resolve the discrepancies between the TS allowable drop time and the measured drop times, Waterford 3 and Combustion Engineering reviewed the affected safety analyses to support increasing the allowable drop time to 3.2 seconds. This review, and the following discussions, incorporate the same methodology recently utilized by Arkansas Nuclear One Unit 2 in successfully presenting a similar TS change to the NRC.

First, 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 recent testing, for example, demonstrated that the majority of the CEAs were inserted beyond 90% at 3.0 seconds and many were actually fully inserted. As a result, it is apparent that the total reactivity insertion remains greater than the safety analyses assumption since the "early" CEAs more than offset the CEAs which do not meet the technical specification criteria.

It is noteworthy that many of the existing design basis 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 than allowed. Nonetheless, the following discussion credits none of these conditions, demonstrating the significant conservatisms in the analyses.

Waterford 3 design basis accidents were re-evaluated. In light of the increased rod drop time, the accidents were 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 & 3	4	Uncontrolled CEA withdrawal (at low power and power)
15.4.1.4	1	CEA misoperation
15.4.1.5	2	CVCS malfunction (inadvertent boron dilution)
15.4.1.6	1	Startup of an inactive reactor coolant system pump
15.4.1.7	2	Uncontrolled CEA Withdrawal from a subcritical condition
15.5.1.1	2	CVCS malfunction
15.5.1.2	1	Inadvertent operation of the ECCS during power operation
15.9.1	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]

(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.

FSAR Subsection	Category	Event
		LIMITING FAULTS
15.1.3.1	2	Steam system piping failures
15.1.3.2	2	Steam system piping failures - modes 3 and 4, Full Length CEAs Inserted
15.1.3.3	4	Steam system piping failures - Pre-Trip Power Excursion Analysis
15.2.3.1	4	Feedwater system pipe breaks
15.2.3.2	2	Loss of Normal Feedwater Flow with an Active Failure in the Steam Bypass System
15.3.3.1	4	Single Reactor Coolant Pump (RCP) Shaft Seizure
15.3.3.2	4	Single RCP Shaft Seizure with a Stuck Open Secondary Safety Valve
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 & 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	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

Key to Categories:

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. Therefore, a more detailed discussion is provided.

Those events potentially impacted by a 0.3 second reactor trip delay are discussed in more detail below.

Potentially Impacted Design Basis Accidents

The following Chapter 15 events involve a rapid approach to a safety limit during the same time frame as the scram. The review of these analyses identified conservatisms which are currently not credited but which may compensate for the increase in CEA drop time.

One conservatism which is credited here 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 conservatisms are identified as noted in the individual event discussions.

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 of reactivity (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.

The results of this event were last reported in support of Cycle 2. In terms of radiological consequences, the limiting Pre-trip power excursion was the outside containment break location. The time of minimum DNBR was 1.9 seconds after the trip breakers open. The transient was modelled 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 exceeding $0.005\%\Delta\rho$. By 1.15 seconds the space time curve is conservative and at the time of minimum DNBR, the delayed space time curve added $.075\%\Delta\rho$ 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). The latest docketed analysis of the feedwater system pipe break includes a loss of normal AC power at the time of trip. Table 15.2-8 of the FSAR shows the sequence of event of the main Feedwater System Pipe Break. It is seen that at 15.4 seconds a High Pressurizer Pressure Trip Condition exists. The trip signal causes the trip breakers to open at 16.5 seconds and CEA motion does not begin until 17.2 seconds. As seen below these assumed analysis values are more than sufficient to cover the proposed increased CEA drop time without invalidating the results. Therefore, the conclusion of this event remain unchanged.

Required by Technical Specifications including the 0.3 second delay before CEA motion

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                         Trip Breakers                  CEA
High pressure <-- 0.9 seconds --> Open --> 0.6 seconds ----> Motion
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Values assumed in FSAR Analysis

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                             Trip Breakers               CEA
High pressurizer <-- 1.1 seconds --> Open --> 0.7 seconds --> Motion
Trip Condition
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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. For Cycle 3 COLSS will preserve this additional Loss of Flow margin. Thus the conclusions of this event are unchanged.

Single Reactor Coolant Pump Shaft Seizure/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. The flow is further reduced by the coastdown of the remaining pumps following the trip. The time of minimum DNBR is determined with the HERMITE code. The flow at the time of DNBR is combined with the initial heat flux valve to determine the fuel performance. The TORC code is used to model hot channel conditions at the minimum DNBR time and fuel failure calculated with the method of statistical convolution. The Seized Rotor event was last presented for cycle 2, and less than 8.5 percent of the fuel pins were reported to have failed. The 8.5 percent value was chosen to bound future results and yet demonstrating acceptable radiological consequences. When cycle 3 specific data is used in conjunction with the increased CEA drop time, the calculated fuel failure remains less than the previously reported results. Therefore, the conclusions of this event are unchanged.

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. Peak RCS pressure occurs 2.5 seconds after the trip breakers open. The latest LOCV analysis used the same static scram curve as the Pre-trip Steam Line break event. Application of the corresponding Space-time scram curve, delayed by an additional 0.3 seconds before CEA motion results in the same comparison as the SLB, see Figure 1. It is seen that the conservatism at the time of peak RCS pressure more than compensates for the insignificant non conservatism discussed in the pre-trip SLB discussion. Therefore, the conclusions of the analysis of this event remain unchanged.

Uncontrolled CEA Withdrawal from Low Power Conditions

An increase of 0.3 seconds prior to CEA motion causes an increase in the peak power of less than 4.0 percent. This increase in core power results in local densities just under the 21 kw/ft SAFDL. A deposited energy calculation is performed as this high local power is for a short duration. A similar calculation was presented to the NRC in the cycle 2 reload analysis report and further explained in response to an NRC question. This calculation indicates that the peak fuel temperature does not exceed 3430°F therefore no melting will occur. Additionally, the DNBR during the transient does not fall below 2.0. Thus, the results of the uncontrolled CEA withdrawal from low power conditions remain acceptable.

Asymmetric Steam Generator Events (ASGT)

The asymmetric steam generator events are examined to determine the degradation in thermal margin. The most severe of these events was the postulated instantaneous closure of a single Main Steam Isolation Valve. With an 18°F differential temperature trip setpoint the ASGT event demonstrated a margin requirement of 112% based on a 2 dimensional Hermite Space time calculation. An increase of 0.3 seconds before CEA movement would result in less than a 2% increase in required thermal margin. Current cycle 3 COLSS margins incorporate this additional margin requirement without change. Therefore, the conclusions of this event are unchanged.

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. Protection against violation of a fuel design limit as a consequence of the excessive heat removal is provided by the low DNBR and high local power density trips. The low steam generator water level trip, high reactor power trip, low steam generator pressure trip, and low pressurizer pressure trip will also serve to protect the plant from exceeding barrier design conditions.

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. This can result in an additional five percent flow from full power conditions.

b) Failure in the Steam Bypass System which could result in an opening of one of the turbine bypass valves. The flowrate of one valve is approximately 10.8 percent of the full power turbine flowrate.

c) An inadvertent opening of an atmospheric dump valve or steam generator safety valve (for a discussion of this occurrence and presentation or results see Subsection 15.1.1.4) caused by operator error or failure within the valve itself. Each atmoshperic dump valve can release approximately 6.2 percent of the full power steam flow. A safety valve will pass approximately 8.4 percent of full power steam flow.

The most severe of these incidents in case B. While the actual increase in steam flow would be approximately 10.8 percent, the increase was assumed to be 13.5 percent, to provide a conservative cooldown rate.

The analyses of increased Main Steam Flow Events used conservative assumptions and bounding input data. A cycle specific analysis would be expected to show that the FSAR analyses remain bounding after including the 0.3 seconds scram delay.

Notwithstanding 1.015 penalty will be applied to the CPC addressable constant BERR1 as discussed below 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.

Increased Main Steam Flow with Loss of Offsite Power

The Increased Main Steam flow with single failure is posutlated to take a 4 pump loss of flow after reaching the DNB SAFDl. Thus the 0.3 second additional delay before CEA motion causes an additional margin degradation due to 0.3 seconds flow coastdown. A power penalty of 1.015 is applied to BERR1 to compensate for the additional flow coastdown. Therefore, the conclusions of this analysis remain 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 analysis of the loss of condenser vacuum analysis presented in section 15.2.1.3. The same compensating conservatisms are present in both analyses.

Shaft Seizure with Stuck Open Valve

This analysis is not affected because the fuel failure evaluated in section 15.3.3.1 is not increased by the increased scram delay.

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).

Trip timing is important to the results of this event. By itself, a 0.3 second trip delay would cause significantly worse results.

The neutronics input to the analysis was chosen to bound future cycles. For cycle 3 operation, the data will be much more benign:

	Docketed	Cycle 3
Reactivity Insertion Rate	1.9×10^{-4} $\Delta\rho$/sec	1.5×10^{-4} $\Delta\rho$/sec
Three Dimensional Peak	7	6

A check case showed that the use of cycle 3 specified data more than compensate for a trip delay of 0.3 seconds.

Uncontrolled CEA Withdrawal from Full 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.

The 0.3 second additional delay before CEA motion would cause more adverse results than those currently docketed. This effect is compensated however by the addition of a 1.015 power penalty on BERR1. Therefore, the conclusions of this event remain unchanged.

CEA Ejection

For this analysis, it is assumed that a complete and instantaneous circumfrential 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, the neutronics input to the anlaysis was chosen to bound future cycles. For cycle 3 operation, the data would be much more benign:

HZP Ejection	Reference Analysis	Cycle 3
Ejected Worth	.825% $\Delta\rho$	.522% $\Delta\rho$
3-D Peak	21.9	16.6

Other Analyses

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 peak containment pressure of 43.7 psig is calculated to occur for a 75% power main steam line break with an initial containment pressure at the maximum Technical Specification allowed value of 1.0 psig. 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 less than approximately 0.2 percent of the total used in the peak pressure analyses. Consequently the resultant peak pressure increase has been estimated to be much less than the existing margin to the containment design pressure limit of 44 psig. As noted above, an increased CEA drop time does not affect the LBLOCA response. Therefore, the conclusions of this analysis are not considered to be significantly affected by the proposed increase in CEA drop time.

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 for Waterford 3 as previously described. 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.

   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 several cases 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.

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 or safety?

   Response: No.

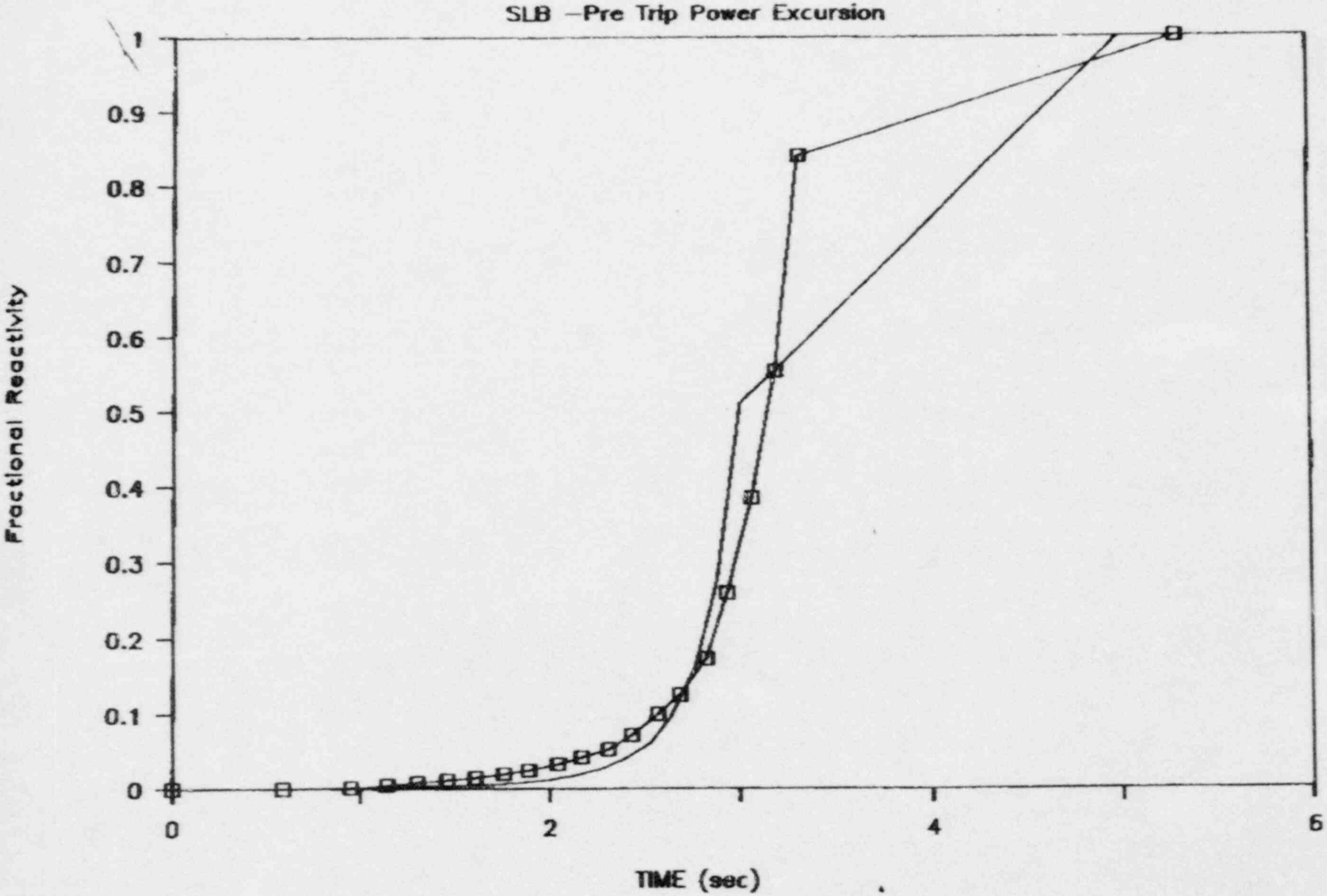
The margins of safety related to CEA insertion are defined by the analyzed events in the Safety Analysis Report which credit their insertion. As demonstrated in response to first question above, evaluation of each affected analysis confirmed that the previously accepted results were either preserved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

Safety and Significant Hazards Determination

Based upon the above Safety Analysis, it is concluded that (1) the proposed change does not constitute a significant hazards consideration as defined by 10CFR 50.92; (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.

FIGURE 1

SLB —Pre Trip Power Excursion



Delayed trip with space-time credit

FSAR assumption

ATTACHMENT A