CRYSTAL RIVER UNIT 3

NUCLEAR GENERATING PLANT

REEVALUATION OF THE MODERATOR DILUTION ACCIDENT

Florida Power Corporation
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REEVALUATION OF THE MODERATOR DILUTION ACCIDENT

Introduction

On February 7, 1977, during routine operations in Mode 5 (Cold Shutdown), Train "A" of the Decay Heat Removal System was being used to recirculate the Reactor Coolant System and the Reactor Coolant System was being filled from the "C" Bleed Tank. In preparation to enter Mode 4 (Hot Shutdown), Reactor Coolant System filling and recirculation was discontinued by closing DHV-5, in order that valves BSV-36 and BSV-37 could be cycled as required by SP-342, "Building Spray System Valve Check."

As BSV-36 was cycled, sodium hydroxide drained by gravity from BST-2 into the Decay Heat System. This gravity draining into the DH System was permitted as DHV-41, DHV-4 and DHV-3 remained open during the cycling of BSV-36, and there was no additional isolation provided between the NaOH Tank and the DH System.

After SP-342 was completed, DHV-5 was opened and recirculation and filling of the RC System was resumed. The NaOH (≈ 600 gallons) which was contained within the suction piping of DHP-3A was then injected into the RC System. The presence of the sodium hydroxide in the DH System and the RC System was not realized until a boron analysis of the RC System was performed.

This operating event was reported to the NRC on March 1, 1977, via LER 77-17, as the chloride and fluoride content of the RC System exceeded Technical Specification 3.4.7 limits due to contaminants contained within the sodium hydroxide added to the RC System. RC System cleanup was initiated via the Makeup and Purification Demineralizer System. The final sodium content of the RC System after cleanup was less than .1 ppm and the final chloride and fluoride content was less than .05 ppm.

The cause of this occurrence was procedural inadequacy in that no precautions in the Surveillance Procedure, SP-342, noted that BSV-36 or BSV-37 should not be cycled with the DH System aligned for operation unless redundant isolation of NaOH Tank BST-2 is provided and the RC System pressure is greater than 12 psig. This pressure in the RC System will insure that swing check valves BSV-153 and BSV-152 remain closed during the cycling of BSV-36 and BSV-37. No hazard was presented to the plant or public as a result of this occurrence.

Florida Power Corporation requested Babcock & Wilcox, the CR#3 NSSS supplier, to analyze this operating event, as it occurred, and also as an unterminated event. These two events were then to be compared with the moderator dilution accident as analyzed in Chapter 14 of the CR#3 Final Safety Analyis Report to determine the bounding event.

On June 9, 1977, B&W responded to FPC's request indicating that:

1. The actual event as it occurred only resulted in small positive reactivity insertion and the core remained sub-critical by a large margin as the dilution event was quickly terminated.

2. The dilution event analyzed considering unterminated dilution would result in the core returning critical, with the time of criticality being determined by the rate of dilution.

3. The Moderator Dilution Accident presented in Chapter 14 of the CR#3 FSAR does not consider the possibility of unterminated dilution. Therefore, the unterminated dilution of the RC System by the contents of NaOH Tank BST-2 is not bounded by the moderator dilution event analyzed in Chapter 14 of the FSAR.

Based on these findings by B&W, FPC filed LER 77-52, identifying this dilution event as an unreviewed safety question and initiated the following reevaluation of the Moderator Dilution Accident at CR#3.

Identification of Cause

The Reactor Building Spray System is designed to furnish building atmosphere cooling to limit post-accident building pressure to less than the design value, and to reduce the building to nearly atmospheric pressure. In addition, sodium hydroxide in the spray enhances removal of the fission product iodine inventory from the containment atmosphere.

The Decay Heat System is designed to maintain core cooling for larger break sizes. This system provides low pressure injection independent of and in addition to the high pressure injection provided by the Makeup and Purification System. In the event of a loss-of-coolant accident, the DH System injects borated water into the reactor vessel for long-term emergency cooling.

The Decay Heat System and the Reactor Building Spray System both take suction on the Borated Water Storage Tank (BWST) within seconds after the initiation of an accident. Sodium hydroxide from one sodium hydroxide storage tank (BST-2) gravity feeds into both systems and mixes with the borated water for pH control in the reactor vessel; sodium hydroxide gravity feeds into the Reactor Building Spray System from the second sodium hydroxide storage tank (BST-1) and mixes with the borated water-sodium hydroxide mixture to maximize

the spray pH before injection into containment. A simpl schematic of the associated piping is shown in Figure I. It is to be understood the system is a modification of the original design in which BST-1 contained sodium thiosulfate for iodine removal.

Both sodium hydroxide tanks are designed and located to permit gravity feeding into the system and are also designed to inject their respective contents at a rate commensurate with the draining rate of the BWST. The contents of each tank are proportioned in such a manner that the correct amount of sodium hydroxide is injected for iodine removal and for pH control.

After the water in the BWST reaches a low level, coincident with the emptying of the sodium hydroxide tanks, the Spray pump and DH pump suction is manually transferred to the Reactor Building sump to recycle discharged fluids, thereby terminating the tank drawdown transient.

The Chemical Additive System, described above, has several alarms, administrative controls, and physical interlocks designed to prevent future improper operation. These are as follows:

1. Prior to aligning the Decay Heat System for operation, manual isolation valves BSV-97 and BSV-98 located in the discharge lines of NaOH Tank BST-2 are closed and the breakers of the motor-operated isolation valves BSV-36 and BSV-37 are

"racked out" with BSV-36 and BSV-37 in the closed position. This provides redundant isolation between NaOH Tank BST-2 and the Decay Heat System. See FSAR Figure 6-3 for the valve diagram discussed above.

2. Verification that the manual isolation valves BSV-97 and BSV-98 are closed and that the pressure of the RC System is greater than 12 psig is performed before the motor-operated valves BSV-36 and BSV-37 are cycled as required by SP-342. When valve cycle testing is performed on BSV-36 and BSV-37, the piping between BSV-97 and BSV-153, and BSV-98 and BSV-152, respectively, is thoroughly flushed with demineralized water to preclude the introduction of NaOH to the Decay Heat System (see FSAR Figure 6-3).

3. The valve cycle testing required by CR#3 Technical Specifications will be performed during a refueling outage.

4. When BSV-36 and BSV-37 are opened, the operator in the Control Room will receive an audible alarm (proposed modification) as well as valve position indicator lights and a tape printout on the events recorder.

5. NaOH Tank BST-2 is equipped with a low level audible alarm in the Control Room which will activate in the event that the tank level drops one foot below its normal operating level. The one-foot drop in tank level corresponds to the draining of approximately 373 gallons of NaOH. In addition to the audible alarm in the Control Room, the operator would receive an annunciator indication and a printout on the events recorder indicating the draining of NaOH Tank BST-2.

It is possible, however, assuming operator error, to establish a flow path from NaOH Tank BST-2 to the Decay Heat System. If it is also assumed that the Decay Heat System is aligned for operation, then dilution of the Reactor Coolant System could occur.

Method of Analysis

The reactor is assumed to be in Mode 5 (Cold Shutdown) or Mode 6 (Refueling) as these are the only modes in which the Decay Heat System is operating. As the time to criticality is dependent upon the dilution flow rate assumed, the maximum dilution flow rate from NaOH Tank BST-2 to the RC System was determined using the Gilbert Associates analytical model.[1]

The GAI analytical model has been verified during the startup and test program of Crystal River Unit #3 and is documented in Appendix G of Reference 1 and in Revision 1 of the CR#3 "Borated Water Storage Tank Drawdown Transient Analysis."[2] The base assumptions and the assumptions used by GAI in each of the four flow case situations are given in Table 1. The results of the GAI analysis for each flow case evaluated are contained in Table 2.

The maximum dilution flow rate calculated by GAI was then used as input by B&W in their reactivity evaluation of an unterminated NaOH dilution event. Assumptions utilized by B&W in their reactivity evaluation for each of the three cases analyzed are given in Table 3. The results of their evaluation for each case are given in Table 4.

Results of Analysis

As can be seen from the results of the B&W evaluation of the unterminated dilution event (Table 4), the minimum time to reach criticality was 22 minutes based on the conservative assumptions of Case I. A plot of subcritical margin versus time for the Case I assumptions is shown on Figure II. The B&W analysis further demonstrates that if the dilution event was to occur during refueling, the entire contents of NaOH Tank

BST-2 could be pumped into the Reactor Coolant System and the reactor would still remain subcritical.

Therefore, the following conclusions are reached as a result of the preceding evaluation:

1. The administrative controls and physical interlocks that have been initiated by FPC are sufficient to preclude any future unplanned addition of NaOH to the Reactor Coolant System of CR#3.

2. The audible alarms and visual indications available to the operator in the Control Room, taken in conjunction with the length of time (minimum of 22 minutes) required to reach criticality if the dilution event were assumed to occur, enable operator action to terminate the event prior to reaching criticality.

3. The health and safety of the public is in no way compromised by continued operation of Crystal River Unit 3.

BS:hlc

947-P

REFERENCES.

1. Ely, R.F. Jr., "Hydraulic Analysis of Piping Networks Using PIPF Computer Program", Topical Report GAI-TR-105, December, 1976.

2. Ely, R.F. Jr., "Borated Water Storage Tank Drawdown Transient Analysis, Revision 1", for Crystal River Unit 3 Nuclear Generating Plant, Florida Power Corporation, 30 January 1976.

TABLE 1

DILUTION FLOW RATE PARAMETERS

Base Assumptions.

1.	Reactor Coolant System Pressure, psig	0
2.	NaOH Tank BST-2 liquid level elevation, feet	152*
3.	Flow path established from BST-2 to DHP-3A	

Case Assumptions.

Case I

4.	NaOH concentration, weight % NaOH	9.5(1)
5.	DH System flow, gpm	0 (2)
6.	NPSH @ DH pump suction, psia	17.5

Case II

4.	NaOH concentration, weight % NaOH	9.5
5.	DH System flow, gpm	3,000 (3)
6.	NPSH @ DH pump suction, psia	22 (≈14 ft. H_2O)

Case III

4.	NaOH concentration, weight % NaOH	20.0(1)
5.	DH System flow, gpm	0 (2)
6.	NPSH @ DH pump suction, psia	15

Case IV

4.	NaOH concentration, weight % NaOH	20.0
5.	DH System flow, gpm	3,000 (3)
6.	NPSH @ DH pump suction, psia	25 (≈14 ft. H_2O)

NOTES TO TABLE 1

(1) The two concentrations of NaOH assumed were 20%, nominally the concentration per present technical specifications, and 9.5%, nominally the concentration per proposed revisions to the technical specifications.

(2) No normal DH System flow, DH pump suction on BST-2 only.

(3) DH System operating normally, recirculating 3,000 gpm with the RC System. BST-2 aligned to feed into DH pump suction also.

* Plant datum elevation is 119 ft. NaOH Tank BST-2 liquid level is 33 ft.

TABLE 2

DILUTION FLOW RATE RESULTS

Case I	
Maximum NaOH Flow Rate, gpm	350
Case II	
Maximum NaOH Flow Rate, gpm	325
Case III	
Maximum NaOH Flow Rate, gpm	350
Case IV	
Maximum NaOH Flow Rate, gpm	300

TABLE 3

MODERATOR DILUTION ACCIDENT PARAMETERS

Case I

Assumptions

(1) Initial boron concentration 1488 ppm*
(2) Reactor Vessel depressurized at $T < 100°F$
(3) Reactor Vessel volume drained to approximately RV outlet nozzle height which corresponds to $\simeq 3000\ ft^3$ of liquid.
(4) All CRA's in (one stuck rod)
(5) BOL conditions
(6) NaOH assumed to be demineralized water for the purposes of moderator dilution
(7) Dilution flow rate assumed to be 350 gpm as calculated by Gilbert Associates
(8) Boron reactivity worth, 75 ppm/1% $\Delta k/k$

Case II

Assumptions:

(1) Initial boron concentration 1950 ppm**
(2) Reactor Vessel depressurized at $T < 100°F$.
(3) Full reactor vessel volume ($\sim$4058 ft3) as would be prior to head removal for refueling.
(4) All CRA's in (one stuck rod).
(5) BOL conditions
(6) NaOH assumed to be demineralized water.
(7) Dilution flow rate assumed to be 350 gpm.
(8) Boron reactivity worth, 75 ppm/1% $\Delta k/k$

Case III

Assumptions

(1) Initial subcriticality of 5% $\Delta k/k$ and ppm boron based on all CRA's in (one assumed stuck rod).
(2) Reactor Vessel depressurized at $T < 100°F$.
(3) Full reactor vessel volume (4058 ft^3) as would be prior to head removal for refueling
(4) NaOH assumed to be demineralized water.
(5) Dilution flow rate assumed to be 350 gpm.
(6) Boron reactivity worth, 75 ppm/1% $\Delta k/k$

* Boron concentration of R.C. System for actual dilution event reported in LER-77-17.
** Technical Specification Mode 5 condition prior to entering Mode 6, refueling.

TABLE 4

MODERATOR DILUTION ACCIDENT RESULTS

Case I

Results

Time to reach criticality was calculated to be 22 minutes; emptying 7700 gallons from the NaOH Tank.

Case II

Results

The amount of dilutant required to go critical exceeds the 12,300 gallon capacity of the NaOH tank, therefore, the core would not return critical under these conditions and assumptions.

Assuming an unlimited source of dilutant, it would require 55 minutes at 350 gpm to return critical.

Case III

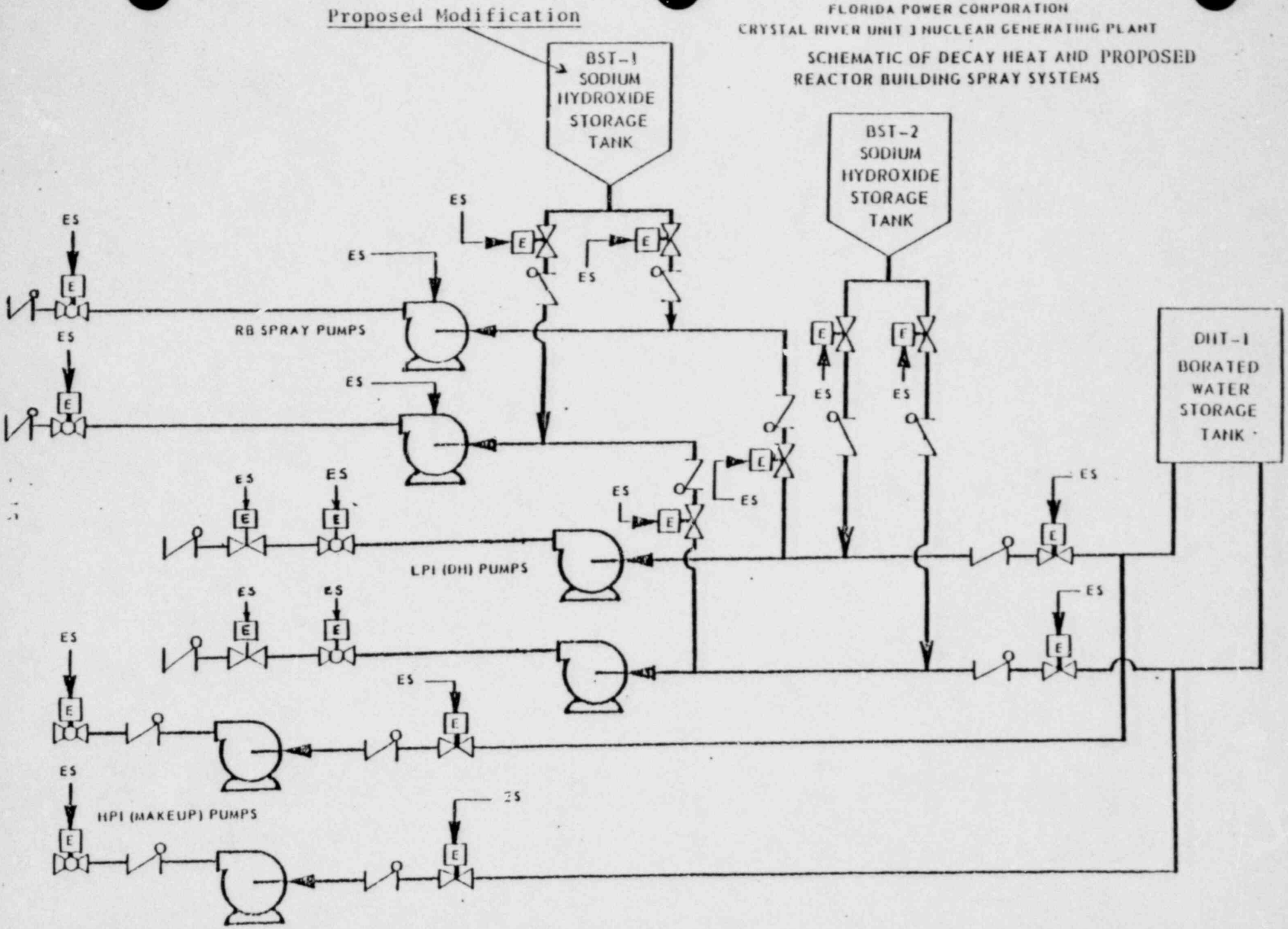
Results

The time required to reach criticality was calculated to be 27 minutes; emptying approximately 3/4 of the NaOH tank.

FIGURE I

FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

SCHEMATIC OF DECAY HEAT AND PROPOSED REACTOR BUILDING SPRAY SYSTEMS



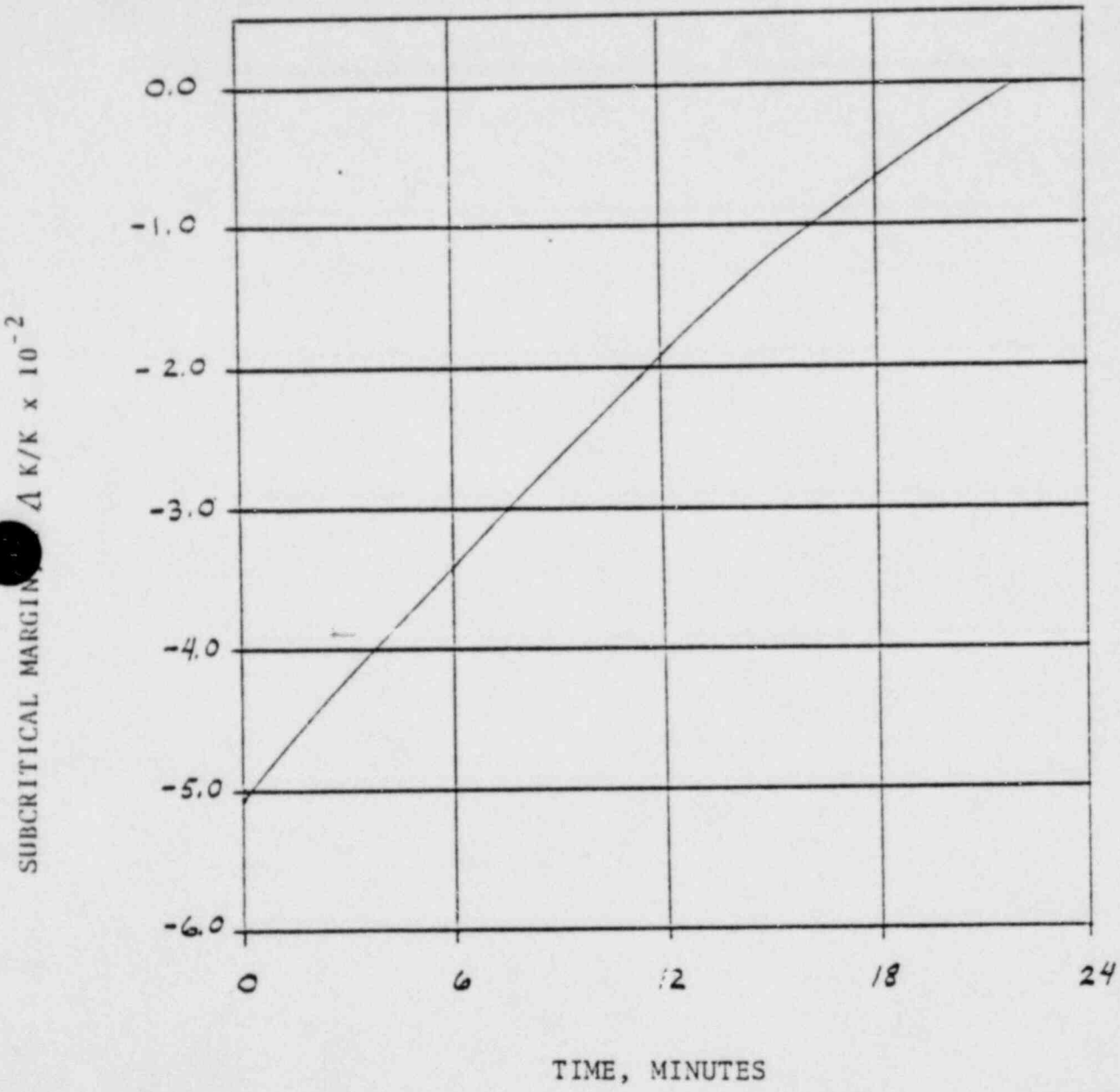


FIGURE II

Subcritical Margin Versus Time for Case I Assumptions