1. OLCR-NLS-85/06 (Supplemental Submittal)

SUBJECT: Facility Operating License NPF-29, page 9.

DESCRIPTION: It is proposed to change Operating License Condition 2.C.(28) to read as follows:

> MP&L shall have on its nuclear operations staff, one or more corporate management officials or advisors (who may be either permanent employees or contracted consultants) who have substantial commercial nuclear power plant operating management experience and who will advise on all decisions affecting safe operation of the plant. This requirement shall be in effect until the plant has accumulated at least 6 months at power levels above 90% of full power.

JUSTIFICATION: On November 14, 1985 MP&L submitted proposed changes to the GGNS Unit 1 Operating License. As part of those changes, it was proposed to delete Operating License Condition 2.C.(28), Advisor to Vice President.

License Condition 2.C.(28) of Facility Operating License NPF-29 requires MP&L to provide an advisor to the Vice President, Nuclear Operations until the plant has accumulated for at least six months at power levels above 90% of full power. This license condition was imposed in June 1982, due to the lack of operating plant experience by corporate management. In March 1985, MP&L filled the position Vice President, Nuclear Operations with Mr. O. D. Kingsley, Jr. who has significant, responsible operational experience.

Mr. Kingsley is a graduate of Auburn University with a B. S. degree in Engineering Physics. He served in various line-officer capacities in the nuclear submarine service for five years before being employed by Alabama Power Company. The following are some of the positions held during his 14 year tenure there.

- Senior Engineer
- Assistant Plant Manager, Farley Nuclear Plant
- Plant Manager, Farley Nuclear Plant
- Assistant Manager - Nuclear Generation
- Manager, Nuclear Engineering and Technical Support
- Director, Nuclear Plant Support - Southern Company

In addition to the commercial nuclear power experience of Mr. O. D. Kingsley, Jr., the President and Chief Operating Officer of MP&L, Mr. W. Cavanaugh, III, has extensive commercial nuclear power experience.

Prior to assuming his present position with MP&L in early 1984, Mr. Cavanaugh worked for Arkansas Power and Light Company for 15 years. He was employed by that company in April, 1969, and worked on the design, construction, licensing, and operation of Arkansas Nuclear One Units 1 and 2. The following are some of the positions he subsequently held during his tenure there:

Assistant Plant Superintendent of Arkansas Nuclear One
Production Project Manager of Arkansas Nuclear One
Manager of Nuclear Services
Assistant Director of Power Production
Executive Director, Generation and Construction
Vice President, Generation and Construction
Senior Vice President, Energy Supply.

From April through August 1983, Mr. Cavanaugh was on loan to Louisiana Power and Light Company and served there as a Senior Vice President - Nuclear Operations with direct responsibility for Waterford 3. He is a graduate of Tulane University with a Bachelor of Science Degree in Mechanical Engineering and served as an officer in the U. S. Naval Submarine Nuclear Program prior to employment with Arkansas Power and Light.

MP&L has had contractors fill the position of Advisor to the position of Vice President, Nuclear Operations for over three years with approximately 65 days of that time while the plant was above 90% of full power (first averaged greater than 90% for a full day on May 12, 1985). MP&L requested deletion of License Condition 2.C(28) in light of the extensive commercial nuclear power experience of its existing Vice President, Nuclear Operations and its President and Chief Operating Officer in a letter to the NRC dated November 11, 1985 (AECM-85/0360).

Based on subsequent conversations between the NRC staff and MP&L, MP&L now proposes that Operating License Condition 2.C.(28) not be deleted but reworded to allow credit to be taken for the present MP&L management experience.

This proposal supplements and modifies that part of the previously proposed change to the Operating License (submitted on November 14, 1985) dealing with Operating License Condition 2.C.(28). A notice on the significant hazards considerations of the previously proposed change was published in the Federal Register on December 3, 1985. Because the intent and content of this proposed modification are essentially unchanged with regard to significant hazards considerations, MP&L believes the December 3, 1985 Federal Register notice suffices to meet the requirements of 10CFR50.91.

SIGNIFICANT HAZARDS CONSIDERATIONS:

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because there is no significant change in the intent of the license condition since the underlying requirement of assuring that corporate management has an appropriate level of operating experience or is advised by persons who do until that experience is attained.

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated because the change in no way affects the design or the procedures involving day to day operation of the plant.

The proposed change does not involve a significant reduction in the margin of safety because the change is administrative in nature.

Therefore, the proposed change involves no significant hazards considerations.

Submitted as PCOL-85/19
November 1, 1985

(b) Final evaluations and recommendations from the TDI Owners Group Program applicable to GGNS Unit 1, and MP&L's actions in response to this program for the standby diesel generators shall be submitted for NRC review and approval prior to startup following the first refueling outage.

(26) Turbine Disc Integrity (Section 10.2.1, SER, SSER #1)

Prior to exceeding 50,000 hours of operation ~~During each refueling outage~~ MP&L shall ultrasonically inspect the bores and keyways of the low pressure turbine discs for indications of cracking. All unacceptable indications and their dispositions shall be reported prior to startup for the next cycle of operation. These inspections shall continue on a 50,000 hour interval until the potential for turbine disc cracking has been assessed and an acceptable alternate inspection schedule has been established.

(27) Circulating Water System (Section 10.4.5, SER)

MP&L shall not fill the Unit 2 circulating water system (including the natural draft cooling tower basin) until Unit 1 flooding concerns related to this system are resolved to the satisfaction of the NRC staff.

(28) Advisor to Vice President (Section 13.1.1, SER, SSER #2, SSER #4, SSER #5)

INSERT

~~MP&L shall provide one or more additional staff members, reporting directly to a Vice President principally in charge of nuclear operations, who have substantial commercial nuclear power plant operating management experience and who will act as advisors to the vice president on all decisions affecting safe operation of the plant. The additional staff members may be permanent employees or contracted consultants, but they shall be retained in this advisory position until the plant has operated for at least 6 months at power levels above 90% of full power.~~

(29) Operating Shift Advisor (Section 13.1.2, SER)

At least one individual on each operating shift shall have substantive previous BWR operating experience, including startup and shutdown of a BWR and under conditions that one might expect to encounter during the initial startup and power escalation at the Grand Gulf plant. This individual is not required to be licensed on Grand Gulf Unit 1 and need not be an MP&L employee, but as a minimum shall be retained on a contract basis to act as a consultant or advisor to the GGNS shift crew. Such an experienced person shall be assigned to each operating shift until the plant achieves and demonstrates full power operation.

INSERT TO OPERATING LICENSE CONDITION 2.C.(28)

MP&L shall have on its nuclear operations staff, one or more corporate management officials or advisors (who may be either permanent employees or contracted consultants) who have substantial commercial nuclear power plant operating mangement experience and who will advise on all decisions affecting safe operation of the plant. This requirement shall be in effect until the plant has accumulated for at least 6 months at power levels above 90% of full power.

2. (NPE-86/04)

SUBJECT: Technical Specifications Table 3.3.7.2-1 and 4.3.7.2-1 pages 3/4 3-64, 65.

DISCUSSION: The proposed change results from a design change to install a Strong Motion Accelerometer (SMA) on a piping support for the injection line to the reactor in the high pressure core spray system. This SMA will provide an acceleration time history for a reactor piping location and confirm the post-seismic evaluation for this plant equipment.

As done on several recent Technical Specification changes involving design changes to the plant, it is requested that the NRC issue the change with an open effective date and require that MP&L notify the NRC within 30 days of the effective date of implementation of the affected technical specification changes. This design change is scheduled for implementation not later than startup following the first refueling outage.

JUSTIFICATION: Currently, there are five (5) Triaxial Time History Accelerographs (SMA's) installed in GGNS Unit 1. One SMA is located on the Unit 1 containment base slab such that it measures the input vibratory motion of the base slab. A second SMA is located in Unit 1 containment attached to the drywell wall at El. 150'-6" on the same containment azimuth as the base slab SMA. A third SMA is located in the Unit 1 auxiliary building attached to one of the standby gas treatment system filter train supports which is seismic Category I equipment. A fourth SMA is located in the standby service water pump house A, which is an independent Category I structure. The fifth SMA is located in the free field approximately 250 feet from any station structure, with axes oriented in the same direction as the containment building accelerometers. All accelerometers, including the one proposed in this submittal, have their principal axes oriented identically, with one horizontal axis parallel to the major horizontal axis assumed in the seismic analysis.

This proposed change results from a design change to add a sixth SMA on a reactor support by restart from the first refueling outage as required by Operating License Condition 2.C.(7). This SMA will be located on a piping support for the injection line to the reactor in the high pressure core spray system. This will allow data collection on a reactor piping support during seismic events.

MP&L believes that the installation of the specified seismic instrumentation in the reactor containment structure along with other Category I structures, systems, and components constitutes an acceptable program as described in the Standard Review Plan (NUREG-0800 Rev. 1 - July 1981) to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the seismic response of major structures and systems. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to allow for post event evaluation. With at least one of the two Seismic Triggers indicating greater than 0.01g acceleration, all Seismic Monitoring System cassette magnetic-tape recorders start and record the Seismic acceleration data from the SMA's. Data obtained will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of the plant are adequate and that allowable stresses are not exceeded under conditions where continuity of operation is intended. Provision of such seismic instrumentation complies with Regulatory Guide 1.12.

SIGNIFICANT HAZARDS CONSIDERATION:

The design change associated with this proposed technical specification change will provide additional assurance that adequate post seismic event data is captured for analysis.

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because it adds required improvements not currently listed in the technical specifications. The purpose of the SMA's is to provide data for post seismic event evaluation and to justify continued plant operation. This instrumentation is not utilized for transient mitigation or detection in the present accident analysis.

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. This change will add a SMA to the high pressure core spray system piping supports, but will in no way affect the operation of that system. The additional SMA provides data only for post seismic event evaluation.

The proposed change does not involve a significant reduction in the margin of safety because the addition of another SMA increases the data collection ability for seismic analysis.

Therefore, the proposed change involves no significant hazards considerations.

TABLE 3.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Strong Motion Accelerometer		
a. Containment foundation	0.001 to 1.0g	1
b. Drywell	0.001 to 1.0g	1
c. SGTS Filter Train	0.001 to 1.0g	1
d. SSW Pump House A	0.001 to 1.0g	1
e. Free Field	0.001 to 1.0g	1
f. Reactor Piping Support	0.001 to 1.0g	1
2. Triaxial Peak Recording Accelerograph		
a. Containment Dome	0.01 to 2g	1
b. Auxiliary Building Foundation	0.01 to 2g	1
c. Diesel Generator 11	0.01 to 2g	1
d. Control Building Foundation	0.01 to 2g	1
e. Control Room	0.01 to 2g	1
f. Reactor Vessel Support	0.01 to 2g	1
g. Reactor Recirc. Piping	0.01 to 2g	1
h. Main Steam Piping	0.01 to 2g	1
i. LPCS Spray Line	0.01 to 2g	1
j. HPCS Spray Line	0.01 to 2g	1
k. SSW Pump House B	0.01 to 2g	1
3. Triaxial Seismic Switches		
a. Containment Foundation (SSE)	0.025 to 0.25g	1*
b. Containment Foundation (OBE)	0.025 to 0.25g	1*
c. Drywell (SSE)	0.025 to 0.25g	1*
d. Drywell (OBE)	0.025 to 0.25g	1*
4. Vertical Seismic Trigger		
a. Containment Foundation	0.005 to 0.05g	1*
5. Horizontal Seismic Trigger		
a. Drywell	0.005 to 0.05g	1*

*With control room annunciation.

TABLE 4.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Triaxial Strong Motion Accelerometer			
a. Containment Foundation	M	SA	R
b. Drywell	M	SA	R
c. SGTS Filter Train	M	SA	R
d. SSW Pump House A	M	SA	R
e. Free Field	M	SA	R
f. Reactor Piping Support	M	SA	R
2. Triaxial Peak Recording Accelerograph			
a. Containment Dome	NA	NA	R
b. Auxiliary Building Foundation	NA	NA	R
c. Diesel Generator 11	NA	NA	R
d. Control Building Foundation	NA	NA	R
e. Control Room	NA	NA	R
f. Reactor Vessel Support	NA	NA	R
g. Reactor Recirc. Piping	NA	NA	R
h. Main Steam Piping	NA	NA	R
i. LPCS Spray Line	NA	NA	R
j. HPCS Spray Line	NA	NA	R
k. SSW Pump House B	NA	NA	R
3. Triaxial Seismic Switches			
a. Containment Foundation (SSE)	M	SA	R
b. Containment Foundation (OBE)	M	SA	R
c. Drywell (SSE)	M	SA	R
d. Drywell (OBE)	M	SA	R
4. Vertical Seismic Trigger			
a. Containment Foundation	M	SA	R
5. Horizontal Seismic Trigger			
a. Drywell	M	SA	R

3. (NPE-86/05)

SUBJECT: Technical Specifications Tables 3.3.3-1, 3.3.3-2, and 4.3.3.1-1; pages 3/4 3-28, -31, -34, and -35.

DISCUSSION: The proposed change to Technical Specifications Tables 3.3.3-1, 3.3.3-2 and 4.3.3.1-1 adds instrumentation, setpoints and surveillance requirements for ADS actuation instrumentation, and changes the name of the currently identified ADS Timer to ADS Initiation Timer. The instrumentation to be added to these tables are ADS Bypass Timer (High Drywell Pressure) and Manual Inhibit switches. These changes to the Technical Specifications are needed to supplement a design change required by Operating License Condition 2.C.(33)(f).

Mississippi Power & Light Company (MP&L) is now in the process of developing a design change to satisfy the license condition. This design change is scheduled for implementation not later than startup following the first refueling outage. As with several recent Technical Specification changes involving design changes to the plant, it is requested that the NRC issue the change with an open effective date and require that MP&L notify the NRC within 30 days of the effective date of implementation of the affected technical specification changes.

Operating License Condition 2.C.(33)(f) requires MP&L to modify the Automatic Depressurization System (ADS) logic in accordance with Option 4 of the BWR Owners' Group Evaluation of NUREG-0737, Item II.K.3.18, prior to startup following the first refueling outage. The Option 4 modification is the addition of a timer that bypasses the high drywell pressure permissive if the reactor water level is low for a sustained period and changes the low reactor pressure vessel (RPV) water level trip set point to the top of the active fuel. No change to the low-low-low water level trip is proposed because the trip is provided by the wide range instrumentation and the lowest level measured by this instrumentation is 373 inches above vessel zero. As shown in Bases Figure B 3/4 3-1 of the Technical Specifications, the level at the top of the active fuel is 366.3 inches above vessel zero. The RPV water level 1 trip setpoint is 382.7 inches above vessel zero.

In addition to the modifications, the License Condition requires MP&L to provide justification for the timer delay settings, revise the emergency procedures for use of the manual inhibit switch, and submit proposed Technical Specification surveillance procedures for the timer and switch. The modifications and justification for the timer delay settings are addressed below. The emergency procedures for use of the manual inhibit switch will be revised and available for NRC review and concurrence prior to completion of the modifications. The proposed technical specifications surveillance requirements are addressed with this submittal.

Through selected safety/relief valves, the ADS functions as a backup to the operation of the high-pressure coolant systems. The ADS depressurizes the vessel so that low-pressure systems may inject water into the reactor vessel. ADS is activated automatically upon coincident signals of low water levels (Level 1 & 3) in the reactor vessel and high drywell pressure. In addition, a low-pressure emergency core cooling system pump running permissive must be satisfied prior to ADS initiation. A time delay of approximately 105 seconds after receipt of the coincident low water level and high drywell pressure signals allows time for the automatic blowdown to be reset manually if the operator believes the signals are erroneous or if the water level can be restored. For transient and accident events that do not directly produce a high drywell pressure signal (e.g., stuck-open relief valve or steam line break outside containment) and are degraded by a loss of high-pressure coolant systems, manual initiation of the ADS is required to provide adequate core cooling. The present ADS logic design does not satisfy the criteria of Item II.K.3.18 of NUREG-0737 to eliminate the need for operator action because it has not been demonstrated that the high drywell pressure signal would be present for all situations requiring automatic ADS actuation.

The proposed modification will provide a bypass of the drywell high pressure signal after a set time delay and a manual inhibit function. This will further automate the ADS system by providing automatic ADS initiation, if required, for events such as a break external to the drywell or a stuck open safety relief valve. The modification also provides the capability to more easily inhibit ADS operation in accordance with Emergency Procedure Guidelines (EPGs). The manual inhibit switch allows the operator to inhibit ADS operation without repeatedly pressing the reset pushbutton as is required with the current design. One manual inhibit switch will be provided for each division (1 & 2) of the ADS actuation instrumentation. Each switch will activate a white indicating light and an annunciator to alert the operator of the inhibit action. The pressure relief function and the manual ADS or individual safety relief valve control will not be affected by operation of the manual inhibit switch.

JUSTIFICATION: Modifications: The design modifications required by the License Condition comply with NUREG-0737 by extending automatic initiation of ADS to those transient events which do not result in a release of steam to the drywell but which may require depressurization of the reactor pressure vessel to maintain adequate core cooling. Of the eight design alternatives evaluated by the BWR Owners' Group, the one selected for implementation at GGNS is identified as Option 4. (See Item II.K.3.18 in Supplement 4 to the Safety Evaluation Report). This design alternative bypasses the high drywell pressure portion of the current logic after a specified time interval, and adds a manual switch which allows the operator to inhibit automatic ADS initiation. The high drywell pressure signal is bypassed by the installation of a "bypass" timer which is actuated on low RPV water level - Level 1. (The presently installed timer initiates ADS after receipt of coincident signals as discussed above.) When the high drywell pressure bypass timer times out, the high drywell pressure trip is bypassed and ADS is then initiated on a Level 1 signal alone (coincident with a Level 3 signal and an ECCS pump running and after the 105 second timer runout). The additional logic would not affect the high drywell pressure/low RPV water level initiation sequence insofar as it responds to pipe breaks inside the drywell. It should be noted that once the bypass timer times out, the bypassing of the high drywell pressure would be sealed in and the bypass timer would not automatically reset. The bypass timer is automatically reset if the Level 1 signal clears before the timer times out.

Timer Delay Settings: The delay setting for the bypass timer is a double bounded setpoint in which the lower limit is based on the time needed to allow recovery of RPV water level above Level 1 and minimum SLCS injection time during an Anticipated Transient Without Scram (ATWS) event. The upper limit is based on the avoidance of excessive fuel cladding heatup using 10CFR50, Appendix K models.

The upper analytical limit for the bypass timer delay setting was established based on the limiting ECCS event which does not result in a high drywell pressure trip. A review of the FSAR accident analyses determined that the limiting event is a main steam line break outside of containment assuming a high pressure core spray failure. Using a conservative maximum bypass timer delay setting of 10 minutes, a performance analysis resulted in a calculated fuel element peak cladding temperature (PCT) of 1862°F which is still well below the criteria of 2200°F as given in 10CFR50.46. The attached figures 1 through 4 document the results of the analysis and reflect the effects on system pressure, water level, heat transfer coefficients, and PCT.

For operations after modification of the ADS initiation logic during the first refueling outage until the second refueling outage, operator action is required to manually inhibit automatic ADS initiation under extreme circumstances of a severe ATWS event in accordance with Emergency Procedure Guidelines. After the SLCS modification is implemented during the second refueling outage, this operator action will no longer be necessary.

It should be noted that the lower analytical value for the bypass timer setting takes into consideration the SLCS design as modified for ATWS (i.e., two pump operation and injection of 86 gpm through the HPCS line) and the time for recovery of level 1 using HPCS during the ATWS. The current GGNS design uses a nominal 43 gpm (SLCS) flowrate injected at the bottom of the core. The SLCS modification is scheduled for implementation during the second refueling outage. In the interim, MP&L will evaluate the lower analytical value for the bypass timer setting to be used after the SLCS modification. If the lower limit for the ATWS modification is determined to be necessary, MP&L will propose the necessary changes for implementation prior to startup following the second refueling outage.

With appropriate allowances for loop accuracy, calibration accuracy, and drift, the Technical Specification values for the high drywell pressure bypass timer delay setting were determined to be 9.4 minutes for the maximum allowable value and 9.2 minutes for the nominal trip setpoint.

Attachment 1 to this submittal is an update to the logic diagrams for the Emergency Core Cooling System Actuation Instrumentation reflected in Table 3.3.1-1. These logics were originally submitted in a letter to Mr. Harold R. Denton from Larry F. Dale dated May 8, 1984 (AECM-84/0093).

SIGNIFICANT HAZARDS CONSIDERATIONS:

The proposed amendment does not:

1) involve a significant increase in the probability or consequences of an accident previously evaluated. The limiting accident analysis affected by this change is the steam line break outside of containment assuming HPCS failure. A reanalysis of this event determined that after the water level reaches Level 1 and with the maximum bypass timer setting of 10 minutes, a PCT of 1862°F would occur. This PCT is substantially below the criteria of 2200°F as given in 10CFR50.46. Further, the modifications result in an enhancement of the ADS and do not affect performance of the intended safety function. The modifications extend the automatic initiation of ADS to encompass those transient events which do not produce a high drywell pressure signal but may require depressurization of the reactor vessel to maintain adequate core cooling.

2) create the probability of a new or different kind of accident from any accident previously evaluated. The modifications eliminate the need for operator action for those events which do not produce a high drywell pressure signal but may require ADS actuation. Although operator action may be required for an extremely unlikely ATWS event, the modifications will allow the actions specified in the emergency procedure guidelines to be performed more reliably.

Although new instrumentation will be added for the ADS modification, the instrumentation provides operational enhancements and does not inhibit a valid high drywell pressure signal. The pressure relief function and the manual ADS or individual relief valve control are not affected by operation of the manual inhibit switch. Further, the manual inhibit switch represents an improvement in that the switch allows the operator to inhibit ADS operation without repeatedly pressing the reset button.

3) involve a significant reduction in a margin of safety because margins of safety are not adversely affected. The only margin of safety considered is the peak cladding temperature. The criterion of 2200°F as given in 10CFR50.46 is met with significant conservatism since the PCT for the limiting event is 1862°F. Further, the modifications eliminate the need for operator action for those events which do not produce a high drywell pressure signal but may require ADS actuation. Margins of safety are not affected by the lower limit for the bypass timer delay setting. This limit is determined only by the limiting ATWS event and will be a plant-specific ATWS feature unique to GGNS following ATWS modifications during the second refueling outage.

Therefore, the proposed change does not involve significant hazards considerations.

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
A. DIVISION I TRIP SYSTEM			
1. RHR-A (LPCI MODE) & LPCS SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(b)	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. LPCI Pump A Start Time Delay Relay	1	1, 2, 3, 4*, 5*	31
d. Manual Initiation	1/system(b)	1, 2, 3, 4*, 5*	32
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(b)	1, 2, 3	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. ADS ~~Timer~~ Initiation Timer	1	1, 2, 3	31
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	31
e. LPCS Pump Discharge Pressure-High (Permissive)	2	1, 2, 3	31
f. LPCI Pump A Discharge Pressure-High (Permissive)	2	1, 2, 3	31
g. Manual Initiation	2/system	1, 2, 3	32
h. ADS Bypass Timer (High Drywell Pressure)	2	1, 2, 3	32
i. Manual Inhibit	1	1, 2, 3	32
B. DIVISION 2 TRIP SYSTEM			
1. RHR B & C (LPCI MODE)			
a. Reactor Vessel Water Level - Low, Low Low, Level 1	2(b)	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. LPCI Pump B Start Time Delay Relay	1	1, 2, 3, 4*, 5*	31
d. Manual Initiation	1/system(b)	1, 2, 3, 4*, 5*	32
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(b)	1, 2, 3	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. ADS ~~Timer~~ Initiation Timer	1	1, 2, 3	31
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	31
e. LPCI Pump B and C Discharge Pressure - High (Permissive)	2/pump	1, 2, 3	31
f. Manual Initiation	2/system	1, 2, 3	32
g. ADS Bypass Timer (High Drywell Pressure)	2	1, 2, 3	32
h. Manual Inhibit	1	1, 2, 3	32

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. DIVISION 1 TRIP SYSTEM		
1. RHR-A (LPCI MODE) AND LPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -150.3 inches*	≥ -152.5 inches
b. Drywell Pressure - High	≤ 1.39 psig	≤ 1.44 psig
c. LPCI Pump A Start Time Delay Relay	≤ 5 seconds	≤ 5.25 seconds
d. Manual Initiation	NA	NA
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -150.3 inches*	≥ -152.5 inches
b. Drywell Pressure - High	≤ 1.39 psig	≤ 1.44 psig
c. ADS ~~Timer~~ Initiation Timer	≤ 105 seconds	≤ 117 seconds
d. Reactor Vessel Water Level-Low, Level 3	≥ 11.4 inches*	≥ 10.8 inches
e. LPCS Pump Discharge Pressure-High	145 psig, increasing	125-165 psig, increasing
f. LPCI Pump A Discharge Pressure-High	125 psig, increasing	115-135 psig, increasing
g. Manual Initiation	NA	NA
h. ADS Bypass Timer (High Drywell Pressure)	≤ 9.2 minutes	≤ 9.4 minutes
i. Manual Inhibit	NA	NA
B. DIVISION 2 TRIP SYSTEM		
1. RHR B AND C (LPCI MODE)		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -150.3 inches*	≥ -152.5 inches
b. Drywell Pressure - High	≤ 1.39 psig	≤ 1.44 psig
c. LPCI Pump B Start Time Delay Relay	≤ 5 seconds	≤ 5.25 seconds
d. Manual Initiation	NA	NA
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -150.3 inches*	≥ -152.5 inches
b. Drywell Pressure - High	≤ 1.39 psig	≤ 1.44 psig
c. ADS ~~Timer~~ Initiation Timer	≤ 105 seconds	≤ 117 seconds
d. Reactor Vessel Water Level-Low, Level 3	≥ 11.4 inches*	≥ 10.8 inches
e. LPCI Pump B and C Discharge Pressure-High	125 psig, increasing	115-135 psig, increasing
f. Manual Initiation	NA	NA
g. ADS Bypass Timer (High Drywell Pressure)	≤ 9.2 minutes	≤ 9.4 minutes
h. Manual Inhibit	NA	NA
C. DIVISION 3 TRIP SYSTEM		
1. HPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -41.6 inches*	≥ -43.8 inches
b. Drywell Pressure - High	≤ 1.39 psig	≤ 1.44 psig
c. Reactor Vessel Water Level - High, Level 8	≤ 53.5 inches*	≤ 55.7 inches
d. Condensate Storage Tank Level - Low	≥ 0 inches	≥ -3 inches
e. Suppression Pool Water Level - High	≤ 5.9 inches	≤ 7.0 inches
f. Manual Initiation	NA	NA

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. DIVISION I TRIP SYSTEM				
1. RHR-A (LPCI MODE) AND LPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	M	R(a)	1, 2, 3
c. LPCI Pump A Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
d. Manual Initiation	NA	R(b)	Q	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3
b. Drywell Pressure-High	S	M	R(a)	1, 2, 3
c. ADS ~~Timer~~ *Initiation Timer*	NA	M	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	S	M	R(a)	1, 2, 3
e. LPCS Pump Discharge Pressure-High	S	M	R(a)	1, 2, 3
f. LPCI Pump A Discharge Pressure-High	S	M	R(a)	1, 2, 3
g. Manual Initiation	NA	R(b)	NA	1, 2, 3
h. and i.				
B. DIVISION 2 TRIP SYSTEM				
1. RHR B AND C (LPCI MODE)				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	M	R(a)	1, 2, 3
c. LPCI Pump B Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
d. Manual Initiation	NA	R(b)	Q	1, 2, 3, 4*, 5*
h. ADS Bypass Timer (High Drywell Pressure)	*NA*	*M*	*Q*	*1,2,3*
i. Manual Inhibit	*NA*	*R*	*NA*	*1,2,3*

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
B. DIVISION 2 TRIP SYSTEM (Continued)				
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	$R^{(a)}$	1, 2, 3
b. Drywell Pressure-High	S	M	$R^{(a)}$	1, 2, 3
c. ADS ~~Timer~~ *Initiation Timer*	NA	M	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	S	M	$R^{(a)}$	1, 2, 3
e. LPCI Pump B and C Discharge *g.* Pressure-High	S	M	$R^{(a)}$	1, 2, 3
h. f. Manual Initiation	NA	$R^{(b)}$	NA	1, 2, 3
C. DIVISION 3 TRIP SYSTEM				
1. HPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	$R^{(a)}$	1, 2, 3, 4*, 5*
b. Drywell Pressure-High##	S	M	$R^{(a)}$	1, 2, 3
c. Reactor Vessel Water Level-High, Level 8	S	M	$R^{(a)}$	1, 2, 3, 4*, 5*
d. Condensate Storage Tank Level - Low	S	M	$R^{(a)}$	1, 2, 3, 4*, 5*
e. Suppression Pool Water Level - High	S	M	$R^{(a)}$	1, 2, 3, 4*, 5*
f. Manual Initiation##	NA	$R^{(b)}$	NA	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1. Division 1 and 2				
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	NA	$M^{(e)}$	R	1, 2, 3, 4**, 5**
b. 4.16 kV Bus Undervoltage (BOP Load Shed)	NA	$M^{(e)}$	R	1, 2, 3, 4**, 5**
c. 4.16 kV Bus Undervoltage (Degraded Voltage)	NA	$M^{(e)}$	R	1, 2, 3, 4**, 5**
2. Division 3				
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	NA	NA	R	1, 2, 3, 4**, 5**
g. ADS Bypass Timer (High Drywell Pressure)	*NA*	*M*	*Q*	*1, 2, 3*
h. Manual Inhibit	*NA*	*R*	*NA*	*1, 2, 3*

FIGURE 1

REACTOR VESSEL PRESSURE VERSUS TIME AFTER BREAK, GRAND GULF

MAIN STEAM BREAK (3.5 SQ.FT.), OUTSIDE THE CONTAINMENT, HPCS FAILURE

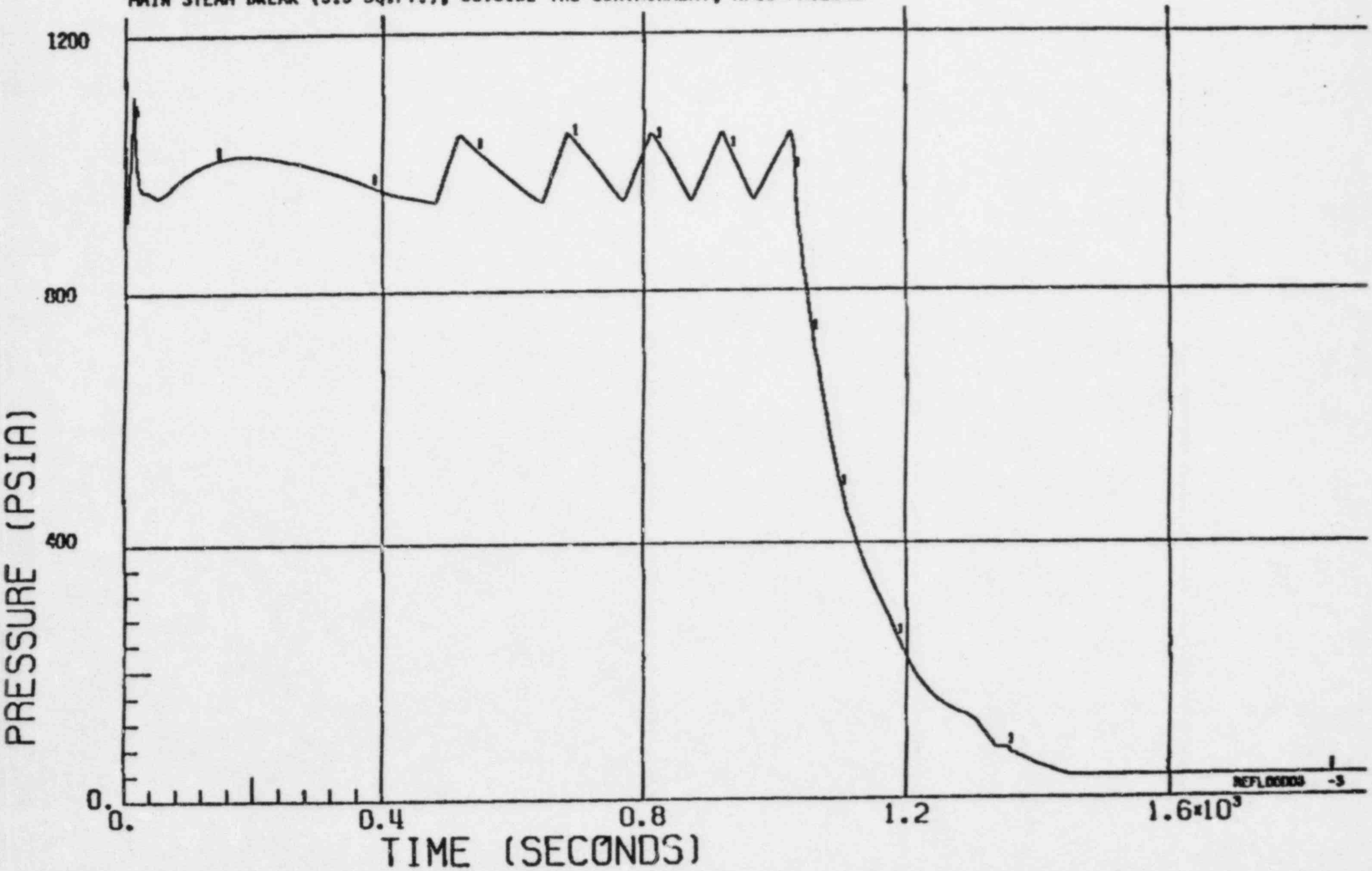


FIGURE 2

WATER LEVEL INSIDE THE SHROUD VERSUS TIME AFTER BREAK, GRAND GULF
MAIN STEAM BREAK (3.5 SQ.FT.), OUTSIDE THE CONTAINMENT, HPCS FAILURE

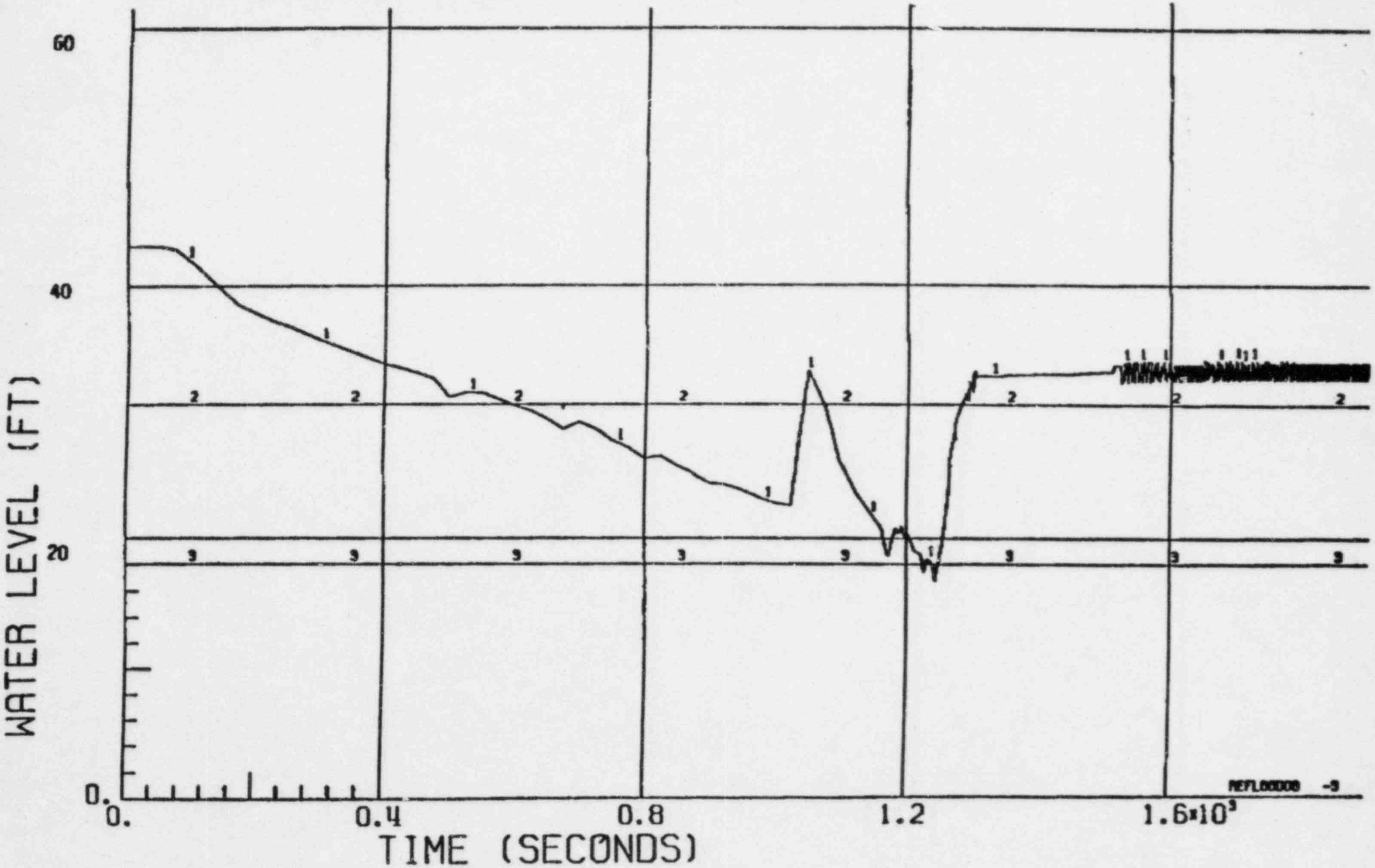


FIGURE 3

PEAK CLAD TEMPERATURE VERSUS TIME AFTER BREAK, GRAND GULF

MAIN STEAMLINE BREAK (3.5 SQ.FT.), OUTSIDE THE CONTAINMENT, HPCS FAILURE

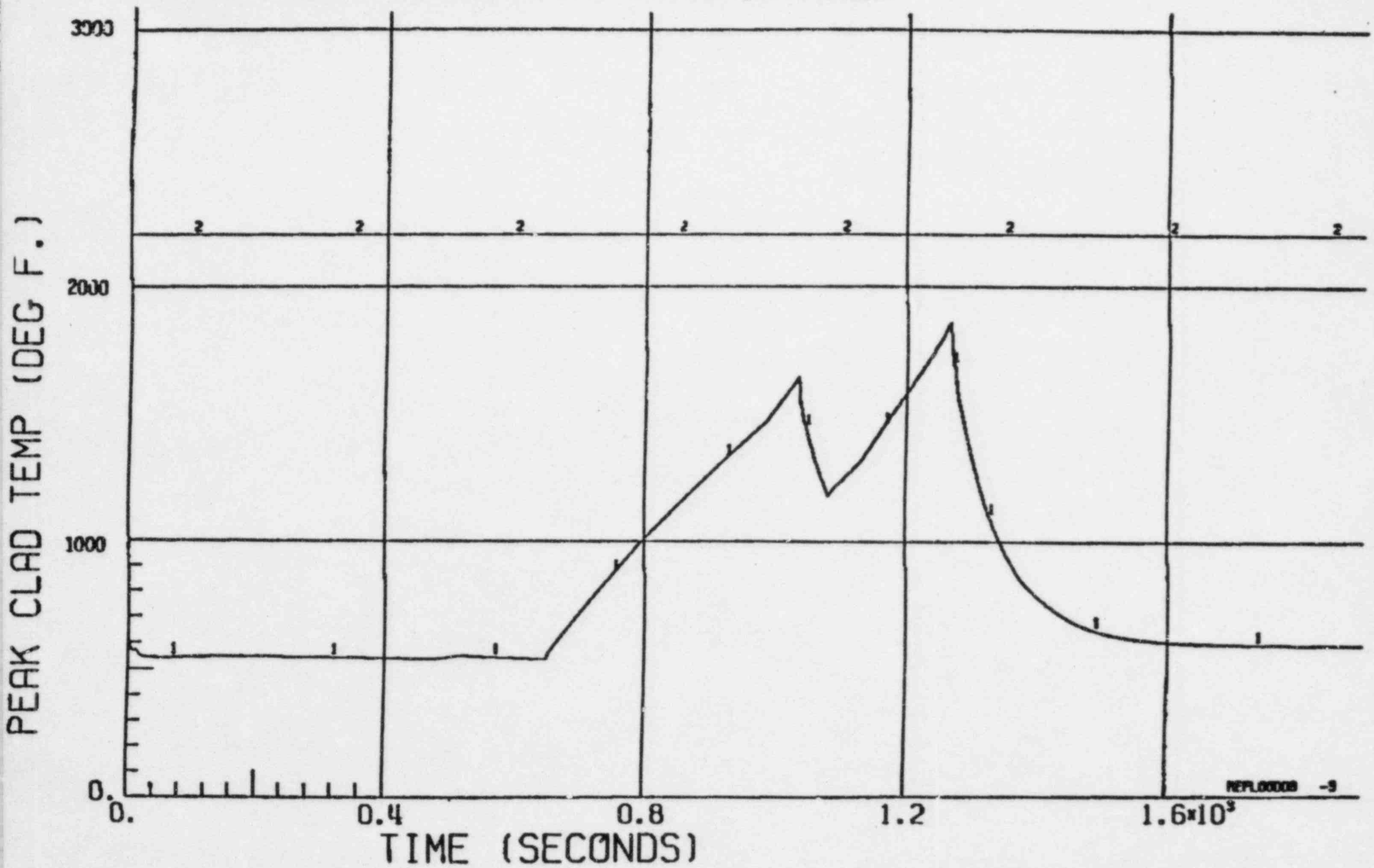
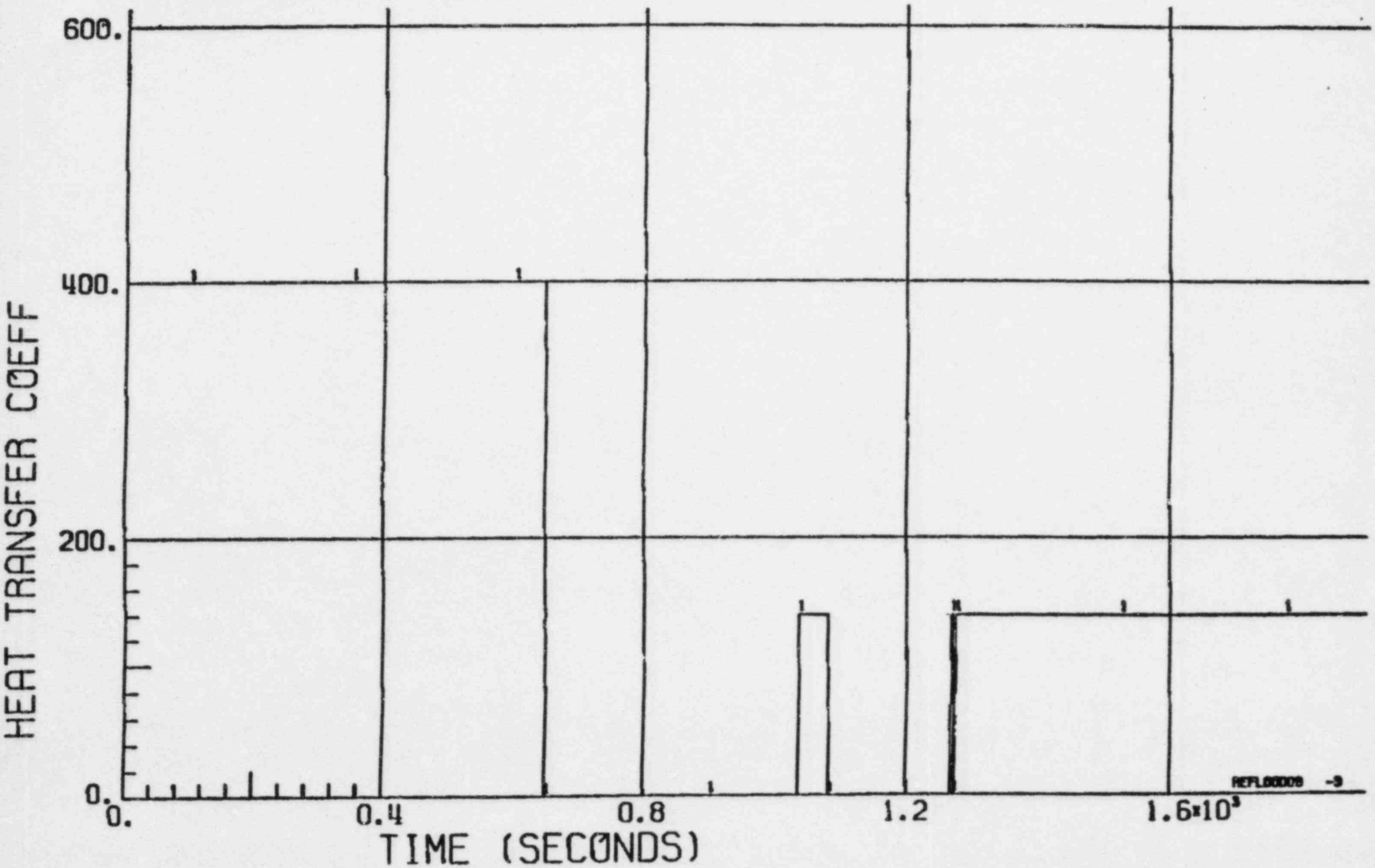


FIGURE 4

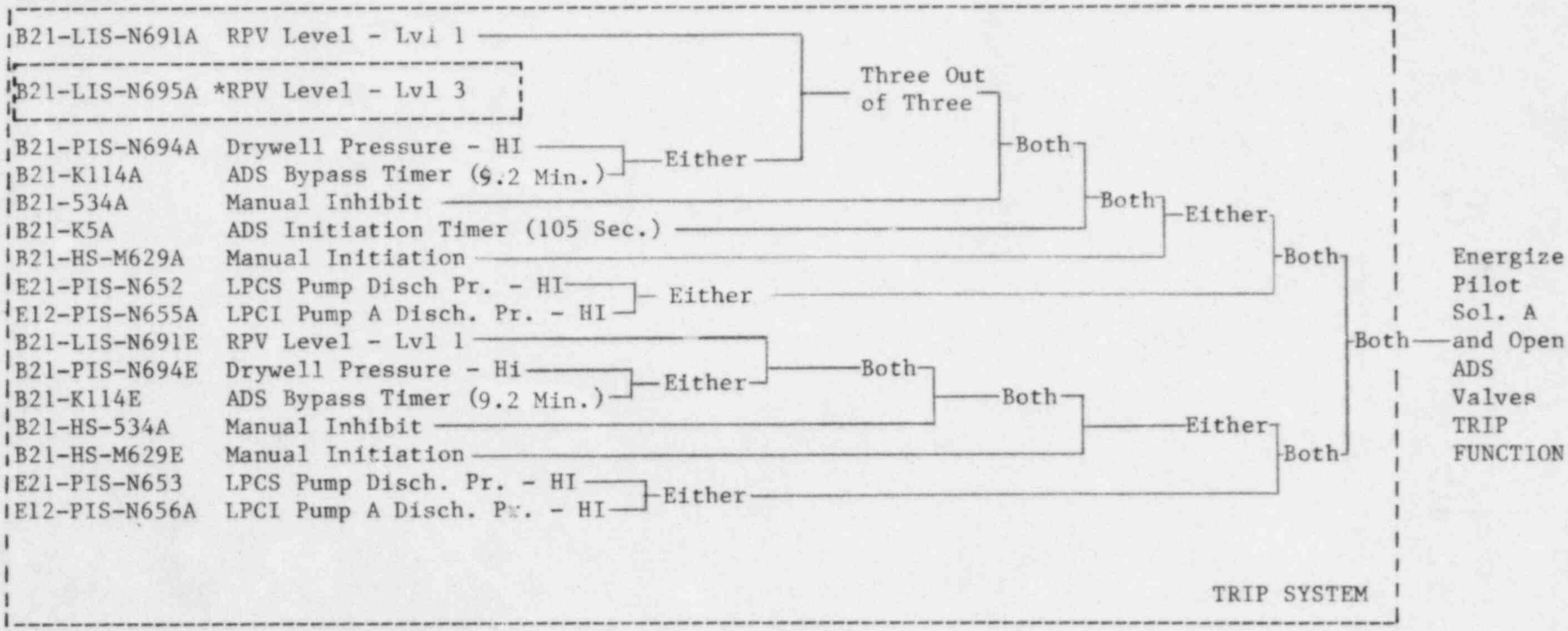
CONVECTIVE HEAT TRANSFER COEFFICIENT VERSUS TIME AFTER BREAK, GRAND GULF MAIN STEAMLINE BREAK (3.5 SQ.FT.), OUTSIDE THE CONTAINMENT, HPCS FAILURE



DEFINITIONS FOR
"CHANNELS", "TRIP SYSTEMS", AND "TRIP FUNCTIONS"
FOR EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION TABLE 3.3.3-1 (Continued)

ADS Trip Systems
ADS "A"

Trip Unit   Parameter   Logic



* One Channel (Typical of 16 shown on this page)

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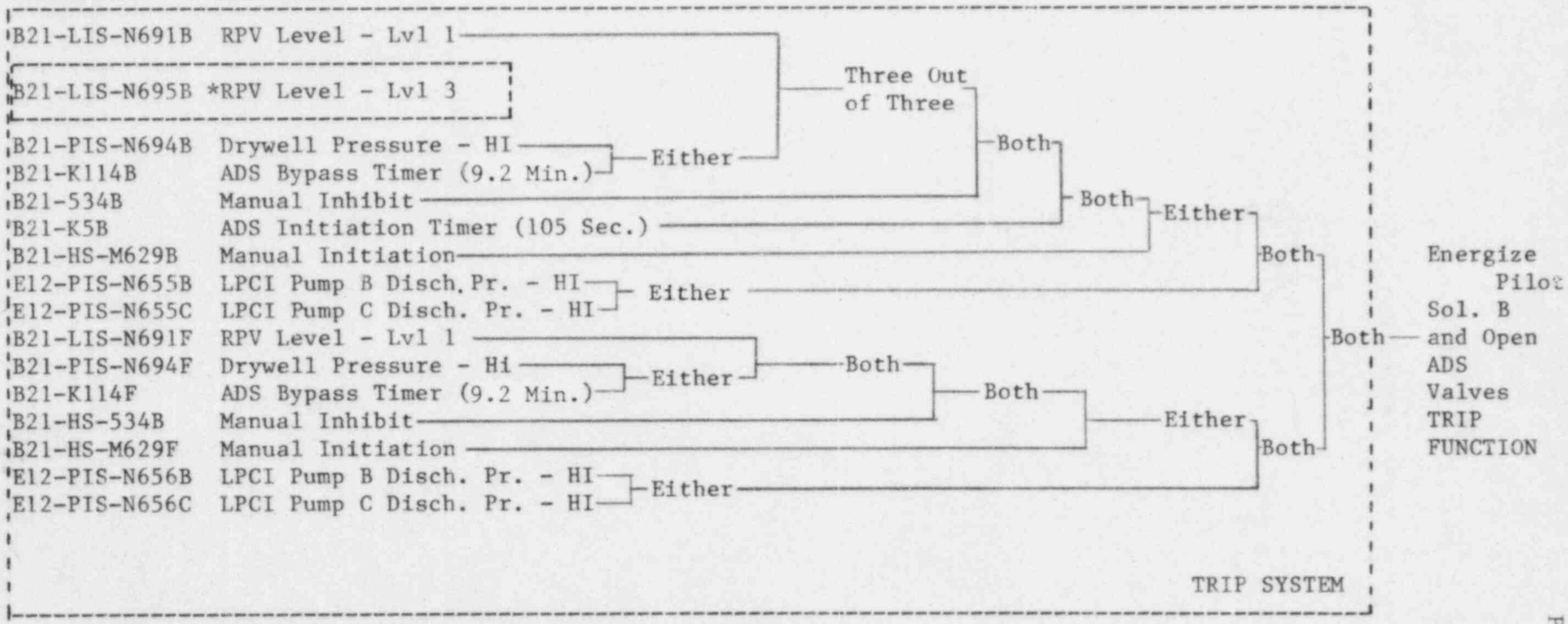
DEFINITIONS FOR
"CHANNELS", "TRIP SYSTEMS", AND "TRIP FUNCTIONS"
FOR EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION TABLE 3.3.3-1 (Continued)

ADS Trip Systems
ADS "B"

Trip Unit | Parameter | Logic



* One Channel (Typical of 16 shown on this page)

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DEFINITIONS FOR
"CHANNELS", "TRIP SYSTEMS", AND "TRIP FUNCTIONS"
FOR EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION TABLE 3.3.3-1 (Continued)

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HPCS System
Trip Unit        Parameter                         Logic

E22-HS-M616      Manual Initiation ---------------------------+
B21-LIS-N673C    RPV Level - Lvl 2 ----+                      |
                                       +-Either-+             |
B21-LIS-N673L    RPV Level - Lvl 2 * --+        +-Both--------+           Initiates
                                                |             |           HPCS and
B21-LIS-N673G    RPV Level - Lvl 2   --+-Either-+             |           Starts
B21-LIS-N673R    RPV Level - Lvl 2   --+                      +---- Any --- Div. 3
B21-PIS-N667C    Drywell Pressure HI --+-Either-+             |     One     Diesel
B21-PIS-N667L    Drywell Pressure HI --+        +-Both--------+
B21-PIS-N667G    Drywell Pressure HI --+        |                         Close HPCS
B21-PIS-N667R    Drywell Pressure HI --+-Either-+                         Pump Disch
B21-LS-N674C     RPV Level - Lvl 8   --+--------- Both -------------------- Valve
B21-LS-N674L     RPV Level - Lvl 8   --+
E22-LIS-N654C    Cond Stg Tk Lvl - Low ------+                            Switch
E22-LIS-N654G    Cond Stg TK Lvl - Low       |                            HPCS
E22-LIS-N655C    Supp. Pool Wtr Lvl - High   +---------------- Any -------- Pump
E22-LIS-N655G    Supp. Pool Wtr Lvl - High --+                 One        Suction
                                                                          from CST
                                                                          to Supp.
                                                                          Pool
                                                                          TRIP
                                                                          FUNCTION
                                                            TRIP SYSTEM
```

* One Channel (Typical of 15 shown on this page)

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