NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
THREE MILE ISLAND, UNIT 1	50-289	1 OF 4

TITLE (4)

ENGINEERING ANALYSIS OF THE LOSS OF 'A'TRAIN DC POWER WITH A LOSS OF OFFSITE POWER AND A LOSS OF COOLANT ACCIDENT

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
07	25	97	97	-- 009 --	00	08	25	97	Crystal River 3	50-302
									FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
POWER LEVEL (10)	100 %	20.2201(b)	20.2203(a)(2)(v)	50.73(a)(2)(i)	50.73(a)(2)(viii)
		20.2203(a)(1)	20.2203(a)(3)(i)	50.73(a)(2)(ii)	50.73(a)(2)(x)
		20.2203(a)(2)(i)	20.2203(a)(3)(ii)	50.73(a)(2)(iii)	73.71
		20.2203(a)(2)(ii)	20.2203(a)(4)	50.73(a)(2)(iv)	OTHER
		20.2203(a)(2)(iii)	50.36(c)(1)	X 50.73(a)(2)(vi)	Specify in Abstract below or in NRC Form 366A
		20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
W. G. HEYSEK, TMI LICENSING ENGINEER	717-948-8191

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On July 25, 1997, as a result of reviewing a plant event reported to the NRC by the Crystal River Unit 3, GPU Nuclear personnel determined that a similar condition existed at TMI-1.

It was identified that a loss of 'A' station DC distribution and a concurrent loss of offsite power would affect both trains of Engineered Safety Features (ESF) equipment. As expected, the 'A' train of ESF equipment would be made inoperable by this event due to the loss 'A' train AC and DC power. In addition, 'B' Train Engineered Safeguards Actuation System (ESAS) would actuate with inability to bypass the "B" train of ESAS. If this failure were to occur concurrently with a large break loss of coolant accident, no procedural guidance was available to operators to allow them to bypass ESAS and take control of components as required to throttle reactor building spray flow and complete switchover to the reactor building sump. The inability to bypass "B" train of ESAS upon loss of power on "A" Train results from a design arrangement in which three analog channels fan out to two trains of logic. Two of the analog channels are powered from "A" Train power. This design has existed since initial plant operation.

The root cause of the condition was determined to be a failure of the plant's designers to properly consider the bypass of Train B of ESAS on a loss of offsite power and a loss of 'A' DC.

Procedure changes were implemented which restored the ability to control the 'B' Train ES components.

The condition was reported per 10 CFR 50.72(b)(2)(iii).

FACILITY NAME (1)	DOCKET NUMBER	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
THREE MILE ISLAND, UNIT 1	50-289	97	-- 009 --	00	2 OF 4

TEXT *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

I. PLANT OPERATING CONDITIONS BEFORE THE EVENT

The plant was operating at 100% power at the time the condition was determined to be reportable and was not changed as a result of that determination.

II. STATUS OF STRUCTURES, COMPONENTS OR SYSTEMS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT.

No systems, structures or components were out-of-service that contributed to the condition.

III. EVENT DESCRIPTION

No event occurred. As the result of a review, a single postulated failure was identified whose consequence had been previously misunderstood.

On July 23rd, 1997, TMI received a fax from Parsons Power group that provided information about a notification made to the NRC on July 21, 1997 at Crystal River-3. Parsons Power (formerly Gilbert Commonwealth) and Framatome (formerly Babcock and Wilcox), designed the Engineered Safeguards Actuation Systems (ESAS) [JE] for both Crystal River-3 and TMI-1. The condition reported had a potential to adversely affect core cooling during a loss of coolant accident (LOCA): as a result of the occurrence of a design basis LOCA with a single failure of 'A' DC distribution [EJ] and a failure of offsite power.

GPU Nuclear personnel reviewed this report, and found that while a different system design exists at TMI-1, the consequences of an 'A' DC distribution failure following a LOCA have similar design deficiencies.

The function of the ESAS is to actuate the Emergency Core Cooling System [BQ] and Reactor Building Isolation and Cooling systems[JM and BK]. These systems mitigate design basis events. LOCA analysis requirements include ECCS functions that must be accomplished with on-site power and a postulated single failure. On July 25, 1997 it was determined that if the assumed single failure were a loss of 'A' DC power, the required manual actions in response to a large break LOCA may not be accomplished in a manner consistent with the safety analysis.

The Vital AC distribution system [ED] is designed to function as the source of regulated battery backup power to the four vital buses for instrumentation and actuation. Vital AC power provides power to the regulated power system (ATA and ATB) [ED] that provides power to indication and plant control systems.

ESAS utilizes a two train, three channel system. Each channel consists of an actuation bistable and a separate bypass permit bistable. The 1A, 1B, and 1C inverters [EF/INVT] power the three channels, respectively. The 'A' actuation train of ESAS receives power from the 1A and 1C inverters, while the 'B' train is powered from the 1B and 1D inverters.

On a concurrent loss of offsite power and 'A' station DC power, the 'A' train emergency diesel generator (EDG) [EK/DG] fails due to loss of field. The resultant 'A' train AC failure and 'A' Train DC failure causes a loss of Inverter 1A and Inverter 1C. The loss of 'A' train AC and DC renders the 'A' train of ESF equipment inoperable. The 'B' EDG starts and loads on the 'B' train safeguards bus. As a result of the losses of the 1A and 1C inverters, two of three channels of engineered safeguards would actuate due to deenergization of the actuation bistables. Engineered safeguards train 'B' would actuate but cannot be bypassed due to deenergization of the bypass permit bistables.

In addition, it was determined that flow indication for the running BS pump is lost due to failure of its power supply. The 'B' train flow indicator, BS1-FI2 [BE/FI] is powered from regulated bus ATA. ATA is ultimately powered from 'A' Train AC and 'A' Train DC.

The ability to bypass ESAS to regain control of ES equipment is required during a large break LOCA to manually throttle BS flow and to manually isolate the BWST after flow is initiated from the reactor building (RB) sump[BE/SUMP]. This action is required to prevent air entrainment in the BS/LPI pumps when taking suction on the RB sump. On a large break LOCA, operators have approximately 40 minutes until the BWST is depleted and suction must be transferred to the RB sump. Revised procedures provide instructions to operators to restore power to the vital bus and prior to isolating the BWST, throttling BS flow and throttling decay heat [BP] flow to maintain adequate pump net positive suction head.

Due to the complexity of this event it could not be assured that operators would be able to restore the ability to bypass by the time these actions are required without procedural guidance in place. Lacking such guidance, GPU Nuclear declared B train LPI and B train BS systems inoperable. A 72 hour Technical Specification limiting condition of operation time clock was entered at 13:30 on July 25 based on that declaration. At 18:55, approved procedural guidance addressing the loss of 'A' DC was made available to the operators and the time clock was terminated.

In addition to the loss of offsite power to the 'A'DC, the effect of the loss of offsite power with the redundant 'B' Train DC was also analyzed and a similar condition was not found to exist.. As a result of offsite power and 'B' station DC power, the 'B' train EDG fails due to loss of field. This 'B' train AC failure causes a loss of Inverter 1B and Inverter 1D. The loss of B train AC and DC power renders the 'B' train of ESAS inoperable. The 'A' EDG starts and loads on the 'A' train safeguards bus. As a result of the loss of the 1B inverter, one of three channels of engineered safeguards would actuate, placing the ESAS in a 1 of 2 channel for trip condition. The remaining two channels would actuate where required, and would be available to bypass if necessary. GPU Nuclear has concluded that because of this specific condition, the 'A' train of ESAS would be operable. The 'B' Train BS flow indicator would not lose power.

The effects of loss of offsite power and loss of a train of DC distribution was also investigated for the other TMI safety actuation systems. This investigation shows no unexpected system response due to this combination of events. For the reactor protection system (RPS) [JC], loss of offsite power alone causes a reactor trip. The loss of two vital buses does not affect RPS performance. For the Heat Sink Protection System (HSPS) [BQ], both main feedwater pumps automatically trip on loss of offsite power. The additional main feedwater valve isolation that occurs due to actuation of two channels of HSPS would have no effect on the heat sink. Two emergency feedwater pumps actuate as designed due to loss of both main feedwater pumps, and loss of all reactor coolant pumps in addition to actuation of two HSPS channels that occurs due to the deenergization of two inverters.

IV. AUTOMATIC OR MANUAL INITIATED SAFETY SYSTEM RESPONSES

Since there was no physical plant event involved with the item being reported herein, there were no safety system responses, automatic or manual.

V. COMPONENT FAILURE DATA

None.

VI. ASSESSMENT OF THE SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

There were no safety consequences associated with the event since no transient condition occurred which would have required changes to the plant configuration or manual or automatic safety system responses. Safety consequences postulated with an actual event resulting from the condition previously described in Section V are the loss of BS flow indication and air entrainment in the BS/LPI pumps due to the loss of net positive suction head.

VII. PREVIOUS EVENTS OF A SIMILAR NATURE

There have been no previous events of a similar nature at Three Mile Island Unit 1.

FACILITY NAME (1)	DOCKET NUMBER	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
THREE MILE ISLAND, UNIT 1	50-289	97	-- 009 --	00	4 OF 4

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

VIII. ROOT CAUSE DETERMINATION

A root cause evaluation was performed by a trained root cause evaluator using NUREG/CR5455. The root cause of the condition was determined to be a failure of the plant's designers to allow the for the bypass of Train B of ESAS on a loss of offsite power and a loss of 'A' DC. This is a system functional design deficiency (INPO Root cause code A1A0 as identified by INPO 86-016). The deficiency was most probably due to designers not properly understanding system interactions when the system was designed in the early 1970s.

Throttling Building Spray flow is an action required when taking suction from the reactor building sump. This requirement was recognized in 1988, and stemmed from the removal of the Sodium Thiosulfate tank in 1980. When the requirement was imposed, it was not recognized that BS flow indication was not powered from the same ESF power train as the BS pump motor.

Failure modes and effects analysis previously performed did not detect the inability to bypass the 'B' train for failure of power in 'A' train. The design reviews include a review by the architect-engineer completed on Oct 12th, 1973, and the ECCS Single Failure Analysis performed by an independent contractor completed on Jan 10th 1977. Both of these design reviews failed to consider the component interactions that would occur and result in the inability to bypass 'B' train ESAS.

IX. CORRECTIVE ACTIONS

Immediate Corrective Actions:
The immediate/completed corrective actions taken in response to the condition were:

1. training operators on the methodology to regain control of the ES equipment and
2. updating Emergency Procedures to reflect the required actions.

The procedure revisions were completed on July 25, 1997 and the on-shift operators were briefed on the event and plant restoration actions within six hours of the discovery of the condition. Each successive crew of operators was provided with the approved procedural guidance for use in restoring power to the 1B channel of ESAS on a concurrent loss of offsite power and a loss of 'A' train DC power. They were instructed in the actions that necessary to allow for the bypass of ESAS to regain control of B train components as previously addressed prior to going on-shift.

Long Term Corrective Actions:
The following actions will be completed within the proposed period.

1. modifying the System Design Description, SDBD-TMI-642 (ESAS), by the end of the fourth quarter 1997.
2. provide engineering support personnel training on the system interactions involved with this event and the need to assure adequate detail is incorporated in design reviews to identify such deficiencies by the end of the first quarter 1998.
3. review and update the "Final Report for TMI-1 Emergency Core Cooling Systems Single Failure Analysis", NSC-LS&R-GPU-0104-1, by the end of the fourth quarter 1997.
4. modifications to the bypass feature of the ESAS and repower the 'B' building spray flow indicator from the ICS Hand subfeed to provide indication will be completed prior to startup from the 13R outage.

* The Energy Industry Identification System (EIIS), System Identification (SI) and Component Function Identification (CFI) Codes are included in brackets, "[SI/CFI]", where applicable, as required by 10 CFR 50.73(b)(2)(ii)(F).