CATEGORY 1

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FACIL:50-261 H.B. Robinson Plant, Unit 2, Carolina Power & Light C 05000261

AUTH.NAME	AUTHOR AFFILIATION
CHERNOFF,H.K.	Carolina Power & Light Co.
MOYER,J.W.	Carolina Power & Light Co.
RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: LER 97-008-00:on 970627,determined that during period of inadequate available NPSH,plant operated outside of design basis.Caused by personnel error.Will modify Safety Injection piping sys to gain addl NPSH margin.W/970728 ltr.

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10 CFR 50.73

CP&L

Carolina Power & Light Company
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville SC 29550

Robinson File No: 13510C
Serial: RNP-RA/97-0162

JUL 28 1997

United States Nuclear Regulatory Commission
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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23
LICENSEE EVENT REPORT NO. 97-08-00

Gentlemen:

The attached Licensee Event Report is submitted in accordance with 10 CFR 50.73. Should you have any questions regarding this matter, please contact Mr. H. K. Chernoff at (803) 857-1437.

Very truly yours,

Dale E Young for

J. W. Moyer
Plant General Manager

Attachment

c: Mr. L. A. Reyes, Regional Administrator, USNRC, Region II
Ms. B. L. Mozafari, USNRC Project Manager, HBRSEP
Mr. B. B. Desai, USNRC Senior Resident Inspector, HBRSEP

IE22

Attachment to Serial: RNP-RA/97-016

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION (4-95) **LICENSEE EVENT REPORT (LER)**	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.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2	50-261	1 OF 7

TITLE (4)

CONDITION OUTSIDE DESIGN BASIS: INADEQUATE SAFETY INJECTION PUMP NET POSITIVE SUCTION HEAD

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
06	27	97	97	-- 08 --	00	07	28	97	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N	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)	X	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)		73.71Appx. G(I)(b)
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		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)
H. K. Chernoff, Supervisor, Licensing/Regulatory Programs	(803) 857-1437

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 DATE (15)	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)

A review of calculations related to the available Net Positive Suction Head (NPSH) for the Safety Injection (SI) system determined that following a postulated large break Loss of Coolant Accident (LOCA), and assuming a single active failure of one SI pump, the remaining operating SI pump might not have sufficient NPSH under certain operating equipment configurations to respond to a design basis accident. The NRC Operations Center was notified of this condition on June 27, 1997, at 1152 hours. This condition was caused by personnel error. Personnel involved in a 1988 design change to the SI system did not adequately assess the impact of a single pump on SI system flow and pump NPSH requirements. Assuming a single failure, failure of the other SI pump could lead to increased fuel cladding temperatures and fuel damage during a LOCA, which could result in an increase in consequences beyond those considered in the current safety analysis. Therefore, during the period of inadequate available NPSH, the plant operated in a condition outside of the design basis. Modifications have been implemented to assure adequate NPSH is available to respond to a design basis event. ECCS flow modeling and calculations will be completed by December 1, 1997, to document the capability of the ECCS system's pumps to perform their design functions. The SI system piping will be modified to gain additional NPSH margin during the next outage of sufficient duration, but prior to startup from Refueling Outage 18. Assessments of other systems will be conducted and will include evaluation of system modifications and related design calculations for potential inadequacies.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2	50-261	97	-- 08 --	00	2 OF 7

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

I. DESCRIPTION OF EVENT

As a result of research in response to questions raised by the NRC during an engineering inspection on April 22, 1997, a review of plant design calculations related to the available Net Positive Suction Head (NPSH) for the Safety Injection (SI) system (EIIS System Code: Q) pumps (EIIS Component Code: P), identified apparent discrepancies between design calculations and the actual configuration and performance of the SI system. Plant engineering personnel promptly began an evaluation of the impact of these discrepancies on the ability of the SI system to fulfill its intended safety function. Pending the outcome of the engineering evaluation, plant personnel also took actions to increase the available water level in the Refueling Water Storage Tank (RWST) (EIIS Component Code; TK), thereby increasing the NPSH available to the SI pumps.

Based on the results of the ongoing engineering evaluation, plant personnel determined on June 27, 1997, that following a postulated large break Loss of Coolant Accident (LOCA), and assuming a single active failure of one SI pump, the remaining operating SI pump might not have sufficient NPSH if two Residual Heat Removal (RHR) pumps and two Containment Spray (CS) pumps were in operation, or if no RHR pumps were in operation. Plant personnel notified the NRC Operations Center on June 27, 1997, at 1152 hours Eastern Daylight Time via the Federal Telephone System (FTS) that the potential existed for this condition to be outside the design basis of the plant. H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2 was operating at 100 percent power at the time of this event.

Background Information

The original SI system design relied upon three SI pumps (i.e., pumps "A", "B", and "C") being available, with at least two pumps delivering flow upon SI initiation. The "A" and "C" SI pumps are powered from separate emergency power trains. In the event of a failure of one of the two emergency power trains, the design provided for the electrical power circuitry to automatically switch the "B" SI pump to the functioning power train. Thus, in the event of a single failure of one emergency power train, the remaining power train would supply power to operate either the "A" and "B" SI pumps, or the "B" and "C" SI pumps.

However, on January 30, 1988, the plant was brought to the cold shutdown condition due to the discovery of a postulated electrical circuitry single failure which could potentially disable two SI pumps. This condition was reported to the NRC on January 28, 1988, and is documented in LER 88-003, revision 1, dated October 24, 1988. In order to correct this condition a plant modification (i.e., Modification M-951, "SI Pump "B" Deletion of Auto Start") was implemented on March 3, 1988. The modification deleted the automatic start feature of the "B" SI pump. Following this modification the "B" SI pump became an installed spare, requiring operator action to manually align power to the "B" SI pump for use. This design modification reduced the available number of SI pumps at the initiation of a SI signal to two. Assuming a single active failure, only one SI pump would then be available for delivery of flow to the core. The 1988 design modification, however, was inappropriately limited in scope to the changes necessary to correct the single failure vulnerability in the electrical system, and did not consider the mechanical impact of operating a single pump on SI system flow and pump NPSH requirements.

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

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2	50-261	97	08	00	3 OF 7

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

I. DESCRIPTION OF EVENT (Continued)

Concurrent with the design modification, LOCA analyses and license amendment actions were conducted to justify plant operation with flow from a single SI pump. On March 7, 1988, Amendment No. 115 to the Technical Specifications (TS) was issued which allowed operation at 60 percent power with automatic start of a single SI pump and manual operator action to start a second SI pump.

The HBRSEP Large Break LOCA/Emergency Core Cooling System (ECCS) Analysis, issued April 25, 1988, assumed one SI pump available following loss of offsite power. The analysis used one half of the flow rate for two SI pumps, which was conservative with regard to the amount of water delivered to the core. The results of the analysis demonstrated satisfaction of the requirements of 10 CFR 50.46(b). This analysis also concluded that sufficient NPSH was available. It should be noted that these activities focused on LOCA analyses which are conducted using degraded pump performance curves and thus NPSH issues at other pump conditions were not identified. On June 20, 1988, Amendment No. 119 to the TS was issued which removed the 60 percent power restriction and allowed return to 100 percent power operations.

Neither the evaluations associated with the 1988 design modification or the subsequent large break LOCA analyses and license amendments adequately addressed the effect of single SI pump operation on the NPSH requirements for the pump. With only one pump running instead of two, the flow rate and resulting flow resistance through the SI system discharge piping is reduced. This results in an increase in the single pump discharge flow rate as compared to the discharge flow rate that would exist if the pump were operating in conjunction with a second SI pump. The increased pump flow rate results in a corresponding increase in the required RWST water level needed to provide adequate NPSH for the pump.

On September 25, 1990, a special test was performed to determine the capability of one SI pump injecting into three cold legs of the Reactor Coolant System. The results of the testing indicated the potential for inadequate NPSH available for either SI pump. This condition was reported to the NRC in LER 90-12 on October 26, 1990. Additional testing was performed, and analysis concluded that the SI pumps could in fact operate properly and perform their safety functions as required. The results of the testing were reported to the NRC in a supplement to LER 90-12 on January 24, 1991. This LER did not provide an adequate analysis of conditions affecting SI pump NPSH. This was documented by plant personnel on March 21, 1991. Identification of analysis deficiencies should have resulted in a revision to LER 90-12-01.

The required RWST level needed to provide adequate NPSH to the SI pumps is dependent on the number of ECCS pumps in operation (e.g., SI pumps, RHR pumps, and Containment Spray (CS) pumps). Calculation RFS-CP-607, "$NPSH_{Av}$ for Pumps in SI System," dated October 2, 1968, performed analyses for three SI pumps running, two CS pumps running, and two RHR pumps running, with RWST level at the tank outlet, and concluded that available NPSH was less than required.

I. DESCRIPTION OF EVENT (Continued)

Calculation RFS-CP-872, "Safety Injection Phase Termination," dated November 12, 1969, performed analyses for three SI pumps running, two CS pumps running, and two RHR pumps running. The calculation concluded that at an RWST level approximately nine feet above ground level elevation, the available NPSH was equal to the required NPSH, and that adequate water had been transferred to the containment for switchover to the recirculation mode where SI pump suction is transferred from the RWST to the containment building sump. This calculation served to establish the basis for the RWST low level setpoint.

Calculation FRSS/SS-CPL-1014, "NPSHA for HHSI and CS at RWST Low-Low Level Alarm," dated June 3, 1987, was performed in support of changes to emergency procedures for continued flow to the core during switchover to recirculation. This provided for continued operation of two SI and one CS pump below the 9 foot level, the RWST low level setpoint, down to RWST low-low level setpoint.

Recent reviews, however, indicate that these early calculations included only minimal detail for the line resistance in the SI piping, and contained inconsistencies with system conditions regarding line flow and configuration. The line loss data from these early calculations was, however, carried through to the subsequent calculations.

Calculation FRSS/SS-CPL-1131, "H.B. Robinson Minimum Safeguards-One SI Pump," dated April 25, 1988, was performed to determine the minimum delivered flow to the Reactor Coolant System from one SI pump specifically for LOCA analysis, but also verified adequate NPSH. This calculation used data from a computer model and 1974 test data. Review of this calculation indicates that the calculation was rigorous with respect to LOCA analysis; however, it does not perform detailed analysis for NPSH. The calculation summary concluded that sufficient NPSH was available for the SI pumps at the low and low-low RWST levels.

The recent engineering review of these calculations concluded that the original design of the SI system was appropriate with regard to the NPSH requirements. However, the 1988 modification did not adequately address the effects of potential operation with a single SI pump. Additionally, the evaluation of the special test performed during 1990 did not adequately assess additional data for SI pump operation at maximum flow rate through three cold legs.

Since investigation of the available NPSH for the SI pumps began in April 1997, the RWST water level has been increased by approximately eleven percent (i.e., 3.5 feet) from the original value. This was accomplished by physically increasing the RWST water level by approximately 9.5 percent and reducing the assumed RWST water level instrumentation setpoint uncertainty by 1.5 percent. These actions were confirmed to be adequate through system flow modeling, and were validated using system performance data from testing performed during 1990. Evaluation of the new calculation data (for which reviews are underway) to date indicates acceptable NPSH is available for SI pumps "A", "B" and "C".

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2	50-261	97	-- 08 --	00	5 OF 7

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

II. CAUSE OF EVENT

This condition was caused by cognitive personnel error. Personnel involved in the design change to support the 1988 system modification to reduce the SI pump requirements from three pumps to two did not adequately assess the impact of a single pump on SI system flow and pump NPSH requirements. A factor which contributed to failure to detect the inadequate NPSH was the unavailability of original calculation details.

Inadequate corrective action in conjunction with LER 90-12, and inadequate design review of Operating Experience information pertinent to this issue prevented earlier detection of this condition. Actions taken as a result of previous events and issues related to NPSH failed to fully assess calculation details.

III. ANALYSIS OF EVENT

Prior to the implementation of a design modification in 1988, system functional design was such that SI pumps "A", "B" and "C" would perform their safety related function. Design changes implemented in 1988, however, reduced the capability of the SI system to perform its safety related function in some scenarios. The investigation revealed that, following the modification, SI pump "B" or "C" running alone in the injection phase would not have adequate NPSH at the original low and low-low RWST level setpoints. The use of a single SI pump instead of two results in the single running pump having a higher flow rate resulting in a higher required NPSH.

Adequate NPSH was available for the small break LOCA and in the recirculation mode for the SI pumps. Only in the event of a single failure of SI pump "A" did NPSH become inadequate, and then only for the large break LOCA case. When emergency procedures were changed to only one SI pump between the low and low-low RWST level for LBLOCA , NPSH become inadequate for SI pumps "B" or "C. SI pumps "B" and "C" would not have had adequate NPSH available during a large break LOCA with the most limiting single failure (i.e., one SI pump running) for the period from the implementation of the design modification on March 3, 1988, until June 27, 1997. Only SI pump "A" had adequate NPSH available during the period. Upon completion of modifications to the RWST on June 27, 1997, SI pump "B" and "C" NPSH was improved and met the NPSH requirement. Current calculations and test data indicate that all three SI pumps have adequate NPSH available to perform their safety function.

Based on inadequate available NPSH and in conjunction with a failure of the "A" SI pump, SI pumps "B" and "C" were inoperable for the period from the implementation of modification M-951 on March 3, 1988, until June 27, 1997. This condition had the potential to result in failure of the operating SI pump during response to a design basis accident. Failure of the SI pump could lead to increased fuel cladding temperatures and fuel damage during a LOCA, which could result in an increase in consequences beyond those considered in the current safety analysis. Therefore, during the period of inadequate NPSH available to two of three SI pumps, the plant operated in a condition outside of the design basis. Accordingly, this report is submitted in accordance with 10 CFR 50.73 (a)(2)(i) as a condition that was outside the design basis of the plant.

IV. CORRECTIVE ACTIONS

Immediate Corrective Actions

Upon discovery that potential discrepancies existed in original design calculations related to the NPSH of the SI pumps, actions were initiated to evaluate the potential effects of the discrepancies and to perform new NPSH calculations. Plant personnel also initiated the following contingency measures to increase the available NPSH for the SI pumps above the minimum required by TS.

On June 7, 1997, a modification was implemented to raise the RWST level by approximately four percent. However, the level was limited by a non-seismically qualified fill line.

On June 9, 1997, the RWST fill line was modified to meet seismic qualifcations, and the RWST level was raised an additional 1.5 percent.

On June 19, 1997, based on preliminary information that indicated that SI pump "C" was potentially most susceptible to potential inadequate NPSH conditions as a result of more restrictive suction lines, plant personnel removed SI pump "C" from service and placed SI pump "B" in service.

On June 25, 1997, the RWST overflow piping was modified, allowing a further increase in the RWST water level. The RWST water level setpoints were revised and the associated instrumentation was re-calibrated to accommodate increased water level.

On June 27, 1997, the RWST level instrumentation was modified, and the RWST water level was raised an additional 5.18 percent.

Planned Corrective Actions

ECCS flow model and calculations for the SI system have been performed and represent an accurate model. When completed, these calculations will document the capability of the SI pumps to perform their design functions, and will be incorporated as the calculation of record by December 1, 1997.

ECCS flow modeling and calculations for the RHR system will be completed by December 1, 1997. These calculations will document the capability of the RHR pumps to perform their design functions.

The SI system piping will be modified to gain additional NPSH margin for SI pumps "B" and "C" during the next outage of sufficient duration, but prior to startup from Refueling Outage 18, currently scheduled for March 1998.

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

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2	50-261	97	-- 08	-- 00	7 OF 7

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

IV. CORRECTIVE ACTIONS (Continued)

A plan will be developed for assessments, modeled after the NRC Architect Engineer Inspection, for other systems, and will include an evaluation of system modifications and related design calculations for potential inadequacies. Design review of pertinent operating experience information will be included in these assessments. An assessment will be conducted during 1998.

This issue will be included as a topic for the Robinson Engineering Support Section (RESS) continuing training during 1997.

LER 91-012-01 will be revised by December 1, 1997.

Previous Actions

The modification process and design control has experienced numerous changes and improvements since 1988. Process oversight improvements and organizational structure changes provide a more positive environment for design control. The original design organization of M-951 was the Nuclear Engineering Department, remotely located at the CP&L Corporate headquarters. The structure of this organization separated engineering reviews by discipline, and was not site specific. As a result, varied priorities and work criteria between groups existed. The present organizational structure is site specific with common goals and management. Also, the design and system engineering staff are located in a common facility providing improved communication and access. The System Engineer reviews modifications to his system, and currently a design review panel reviews each modification. These changes will contribute to the prevention of similar problems. In addition, the current training program for Engineering Support Personnel (ESP) requires each discipline engineer to get a multi-discipline view of the plant and the processes used. This includes system training which discusses NPSH.

V. ADDITIONAL INFORMATION

A. Failed Component Information

None

B. Previous Similar Events

LER 89-010, "Inadequate Auxiliary Feedwater Pump NPSH," identified that inadequate NPSH was discovered based on calculations performed during design basis reconstitution for the Auxiliary Feedwater System.

LER 90-12-01, "Potential of Inadequate NPSH for Safety Injection Pumps," reported the results of a special test was performed to determine the capability of one SI pump injecting into three cold legs of the Reactor Coolant System. The results of the testing indicated the potential for inadequate NPSH available for either SI pump. Subsequent testing was performed, and analysis concluded that the SI pumps could in fact operate properly and perform their safety functions as required.