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FACIL:50-261 H.B. Robinson Plant, Unit 2, Carolina Power & Light C 05000261
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SHEPPARD,J.J. Carolina Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 91-006-00:on 910514,discovered that insufficient analysis existed to ensure adequacy of long-term core cooling in post-LOCA condition.Caused by inadequate calculations.Design basis document completed.W/910610 ltr.

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(10CFR50.73)

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. 91-006-00

Gentlemen:

The enclosed Licensee Event Report (LER), is submitted in accordance with 10 CFR 50.73 and NUREG 1022, Supplements No. 1 and 2.

Very truly yours,

J. J. Sheppard
General Manager
H. B. Robinson S. E. Plant

RDC:lht

Enclosure

cc: Mr. S. D. Ebneter
Mr. L. W. Garner
INPO

JE22

Enclosure to Serial: RNPD[illegible]1-1422

NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2	0 5 0 0 0 2 6 1	1 OF 0 5

TITLE (4)

UNANALYZED CONDITION DUE TO INADEQUATE CALCULATIONS FOR POST-LOCA CORE COOLING

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 NAMES	DOCKET NUMBER(S)
0 5	1 4	9 1	9 1	— 0 0 6 —	0 0	0 6	1 0	9 1		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check one or more of the following)* (11)

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER *(Specify in Abstract below and in Text, NRC Form 366A)*
[ ] 20.405(a)(1)(iii)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[X] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
R. D. Crook, Senior Specialist, Regulatory Compliance	8 0 3 3 8 3 - 1 1 7 9

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES *(If yes, complete EXPECTED SUBMISSION DATE)*   [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT *(Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines)* (16)

On May 14, 1991, a review of details and supporting analysis for the safety injection switchover portion of emergency operation procedures for the H. B. Robinson Plant revealed that insufficient analysis existed for assurance of adequate long term core cooling in the post-LOCA condition. Specifically, previous calculations misinterpreted terminology and therefore did not account for all phenomena depleting the liquid inventory of the reactor core under post LOCA conditions. Thus, the adequacy of procedures for switchover from the injection phase to the long-term cold leg recirculation phase were brought into question. At 1525 hours, the licensee invoked Technical Specification 3.0, and the Plant commenced a power reduction to sixty percent, which was believed to be an analyzed condition. The NRC was notified via the ENS of this condition at 1612 hours pursuant to 10 CFR 50.72(b)(ii)(A).

At 2138 hours, a specific evaluation which justified ninety-five percent power operation was received, and the Technical Specification 3.0 action was exited. Reactor power was increased to ninety percent while additional calculations were performed. Full power operation was justified and achieved on May 29, 1991.

This report is submitted pursuant to 10 CFR 50.73(a)(2)(ii)(A).

NRC Form 366A (9-83) LICENSEE EVENT REPORT (LER) TEXT CONTINUATION U.S. NUCLEAR REGULATORY COMMISSION APPROVED OMB NO. 3150-0104 EXPIRES: 8/31/88

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
H. B. ROBINSON, UNIT NO. 2	0 5 0 0 0 2 6 1	9 1	— 0 0 6	— 0 0	0 2 OF 0 5

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

I. DESCRIPTION OF EVENT

On May 8, 1991, H. B. Robinson Unit No. 2 was operating at one hundred percent power. During review of procedures to support an ongoing Probabilistic Risk Assessment for the Plant, CP&L's Nuclear Engineering Department informally questioned the details and supporting analysis for the safety injection switchover portion of Emergency Operating Procedure EPP-009, "Transfer to Cold Leg Recirculation". Subsequent investigation led to discussion with Westinghouse, the NSSS supplier for H. B. Robinson Unit No. 2[1], and shortcomings of calculations performed during 1989 to support a revision to EPP-009 at that time were confirmed. The licensee was notified of this condition on May 14, 1991. The following is a synopsis of historical information concerning the activities leading up to the discovery of this condition.

In May, 1987, Westinghouse informed CP&L of phenomena (in addition to that of decay heat boiling) depleting the cooling water inventory of the reactor vessel following a Large Break Loss of Coolant Accident (LOCA). Scale testing revealed that loop flow (through the hot leg piping, steam generator, and reactor coolant pump) from the core to a postulated break was much higher than expected. Although not formally explained, the "entrainment", or carry-over of liquid droplets, was a contributing factor in the convective flow. As a result, EPP-009 was revised to minimize the duration and impact of interrupting delivery of cooling flow during switchover from the injection to the recirculation phases.

During 1988, due to the discovery of a original design single-failure discrepancy[2], a Plant modification was implemented to delete the automatic start of the "B" High Head Safety Injection (HHSI) pump. The "B" HHSI pump was then procedurally controlled as a manual start pump. During January, 1989, an evaluation was conducted for the use of only one HHSI pump in the switchover procedure with respect to core cooling. The evaluation compared HHSI delivery to the amount of water boiled by decay heat during the time required to drain the RWST from the twenty seven percent to the nine percent level. This evaluation erroneously used the absence of a closed circuit natural circulation flowpath as justification for considering change in phase (boiling) as the only means of depleting the core water level.

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[1] H. B. Robinson Steam Electric Plant, Unit No. 2, is a Westinghouse Pressurized Water Reactor power plant in commercial operation since March, 1971.

[2] Licensee Event Report 88-003, February 27, 1988.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
H. B. ROBINSON, UNIT NO. 2	0 5 0 0 0 2 6 1	9 1	— 0 0 6 —	0 0	0 3 OF 0 5

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

On May 14, 1991, at 1525 hours, the NRC was notified via the ENS pursuant to 10 CFR 50.72(b)(1)(ii)(A) of an unanalyzed condition, and the Plant began a controlled reduction to sixty percent power. At 2138 hours, the Licensee received a specific evaluation to justify ninety-five percent power operation, and the Plant was brought up to ninety percent load at 0525 hours on May 15, 1991. Additional analysis efforts to support one hundred percent power continued, and at approximately 1250 hours on May 29, 1991, following receipt and review of the additional analysis, the Plant achieved full power.

This report is submitted pursuant to 10 CFR 50.73(a)(2)(ii)(A).

II. CAUSE OF EVENT

The cause of this condition is attributed to personnel error. In evaluating the use of only one HHSI pump in EPP-009, an evaluator underestimated the flow required to maintain core cooling. In focusing on residual or decay heat as the "load", other phenomena in addition to "boil off" were not appropriately addressed. In addition, the Westinghouse letters of May 12, 1987, and June 5, 1987, used terminology consisting of "natural circulation", which were misleading in that it did not explain the phenomena in more detail.

A secondary causal factor that attributed to this condition was that a Westinghouse letter of July 6, 1987, which was submitted to supplement the less detailed letter of June 5, explained the pertinent conditions, phenomena, concerns, and assumptions in detailing the calculational basis. This letter was issued to the Plant, and was not made available to personnel performing the evaluation and verification process in the Nuclear Fuels section at the Corporate office.

III. ANALYSIS OF EVENT

The Safety Injection System is required to provide adequate long term core cooling following a Loss of Coolant Accident. The time period of interest is switchover from injection to recirculation while the RWST level is between the twenty seven percent and nine percent span. The specific concern is whether or not one HHSI pump is adequate to provide long term core cooling, including the three minutes of flow interruption during pump realignment.

This event had no impact on Plant safety. Final analysis demonstrated that full power operation with one HHSI pump available will provide sufficient core cooling in the event of a postulated LOCA. No hardware or Emergency Operation Procedure changes were warranted as a result of this condition.

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
H. B. ROBINSON UNIT NO. 2	0 5 0 0 0 2 6 1	9 1	— 0 0 6	— 0 0	0 4 OF 0 5

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

IV. CORRECTIVE ACTIONS

On May 14, 1991, Westinghouse informed CP&L that the power level at which safe operation could proceed was ninety five percent. This power level was determined by evaluation of the flow from one SI pump from the time of the RWST low level alarm (twenty seven percent) until the time of the low-low level alarm (nine percent). At the time of receipt of the low-low level alarm, the evaluation assumed that one RHR pump is run for one minute to fill the downcomer. After that period of injection, all flow to the RCS is stopped for three minutes while the ECCS system is aligned to take suction from the containment sump. Following the resumption of ECCS flow, one HHSI pump is assumed to inject into the RCS for the remainder of the time for which injection is required. For this evaluation, the core heat source was assumed to be modeled using the ANS 5.1-1979 decay heat plus two sigma.

On May 21, 1991, Westinghouse provided additional information to CP&L concerning the evaluation's utilization of ANS 5.1-1979. Subsequent review of the initial determination indicated that an input value had been incorrectly interpreted to represent ninety-five percent of full power, when in fact it represented one-hundred percent of full power. Therefore, the allowable power, assuming use of the ANS-1979 decay heat plus two sigma, was in fact one hundred percent of full power.

Following that determination, and upon discovery that use of the ANS-1979 model is not consistent with the governing regulations, Westinghouse was requested to determine the power level at which the safe operation of the Plant could proceed using as the core heat source the ANS-1971 decay heat plus twenty percent, as specified in 10 CFR 50, Appendix K. Subsequently, Westinghouse notified CP&L that this power level was determined to be 92.5% of full power, using the same data utilized for the previous analysis.

On May 29, 1991, CP&L was notified that Westinghouse had evaluated the design and operation of H. B. Robinson Unit No. 2 at 100% of full power with one HHSI pump available, and that such operation meets the criteria and specifications of 10 CFR 50.46. For this evaluation, the core heat source was assumed to be modeled using the ANS-1971 decay heat plus twenty percent.

In order to preclude recurrence of this event, CP&L's Nuclear Fuel Section has been placed on Westinghouse distribution for letters to the Plant staff dealing with the subject of accident analysis. In addition, the design basis document for the Safety Injection system has been completed, and is available in the Nuclear Fuels section for use in the design verification process.

NRC Form 366A
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
H. B. ROBINSON, UNIT NO. 2	0 5 0 0 0 2 6 1	9 1	— 0 0 6 —	0 0	0 5 OF 0 5

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

V. ADDITIONAL INFORMATION

A. Failed Components

None.

B. Previous Similar Events

LER-88-003
LER-87-009