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AUTH.NAME AUTHOR AFFILIATION
CROOK,R.D. Carolina Power & Light Co.
PERSON,M.P. Carolina Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 93-018-00:on 931114,Tech Spec violation occurred due to exceeding ramp rate during start-up.Caused by ineffective start-up procedures & operator training.Procedures revised & operators trained.W/931214 ltr.

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Carolina Power & Light Company
Robinson Nuclear Plant
PO Box 790
Hartsville SC 29550

DEC 14 1993

Robinson File No: 13510C
Serial: RNP/93-3105
(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. 93-018-00

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,

Marc P. Pearson
Plant General Manager

RDC:sgk
Enclosure
c: Mr. S. D. Ebneter
Mr. W. T. Orders
INPO

IE22
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Enclosure to Serial: RNP/93-3105

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
H. B. ROBINSON UNIT NO. 2	05000 261	1 OF 6

TITLE (4)
TECH. SPEC. VIOLATION DUE TO EXCEEDING RAMP RATE DURING STARTUP

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 05000
11	14	93	93	-- 018 --	00	12	14	93	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9):
POWER LEVEL (10):

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (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
	20.405(a)(1)(iii)	x	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iv)		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 (Include Area Code)
R. D. Crook, Sr. Specialist-Regulatory Affairs	(803) 383-1179

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)

On November 14, 1993 while returning H. B. Robinson Unit 2 to power operation following Refueling Outage 15, a significant mismatch occurred between actual reactor power and the Power Range Nuclear Instrumentation indicated power level. With the operating crew belief that the net megawatt output of the plant was at 20% power, the actual power was determined by calorimetric to be 30.26%. This constitutes a Technical Specification violation because the power was increased from 20% to 30% at greater than 3% per hour. This event had no adverse impact on safety. The basis for the Technical Specification ramp rate is to minimize the effects of adverse cladding stresses. During this event, the core was maintained in a safe operating condition, and no fuel damage occurred.

This cause of this event is attributed to ineffective startup procedures and operator training. Appropriate procedure revisions and operator training will be completed prior to start-up from the current outage.

This report is submitted pursuant to 10 CFR 50.73(a)(2)(i) as a condition prohibited by the plant's Technical Specifications.

NRC FORM 366A (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
H. B. Robinson, Unit No. 2	505000	93	-- 018 --	00	2 OF 6

I. DESCRIPTION OF EVENT

On November 14, 1993, H. B. Robinson Unit No. 2 was placed on line following startup from Refueling Outage 15. Licensee operators proceeded to stabilize reactor power at an indicated 20% power, with the intention of continuing the power increase at a slower rate to 30%, to perform normal post reactor startup testing consistent with the startup schedule. After the plant was stabilized at 20% power, a licensee manager observing the startup in the control room questioned an apparent discrepancy between indicated power and Net MWe output. Using other diverse indications of reactor power, independent of the Power Range Nuclear Instrumentation (NI), it was determined that the plant was not at 20% power as indicated on the Power Range instruments, but in fact was at approximately 31% power. A calorimetric was performed for an accurate determination of power. The results of the calorimetric confirmed the plant was at 30.26% power. This condition was in violation of Technical Specification 3.10.7.1 because the 3% of rated power per hour limit was exceeded.

The following narrative provides a detailed chronology of the circumstances leading up to this event:

At 0808 hours on November 14, 1993 H. B. Robinson Unit 2 was synchronized with the CP&L grid and placed on line.

At 0833 the auxiliary electrical load was swapped from offsite to on-site power.

At 0842 hours the feedwater controls were placed in automatic.

At 0857 hours the plant was stabilized at Power Range Nuclear Instrumentation indicated 20% power.

At 0922 hours the Technical Support Unit Manager observing the startup in the control room questioned the apparent discrepancy between Power Range Nuclear Instrumentation and the power output in net MWe. The net MWe appeared to be very high for 20% power.

At 1026 a calorimetric was completed in accordance with plant procedure OST-010 which indicated the plant was at 30.26% power. At 1044 the Operations Manager and Technical Support Manager were notified of OST-010 results and power was reduced below 30%.

At 1220 the Regulatory Affairs Section was notified of the Technical Specification 3.10.7 violation for exceeding the 3% per hour ramp rate.

II. CAUSE OF EVENT

Ineffective start-up procedures and operator training are the cause of this event. Each of these inappropriate actions, and contributing causal factors, is discussed as follows:

Control Room personnel did not recognize that the Intermediate Range NIs were indicating 20% Reactor Power while the Power Range NIs were reading 10% Reactor Power. Contributing causal factors were inadequate communication, procedure inadequacies, work practices, and training and qualification of personnel.

Inadequate verbal communication contributed to this event. A pre-job briefing was not performed in accordance with plant procedures, the consequences of potential error were not discussed before starting work, pertinent information was not transmitted and priorities of assigned tasks were not discussed. The pre-job briefing consisted of identifying who was going to do what task and did not include a review of the procedure GP-005, "Power Operation", in sufficient detail. The pre-job briefing did not identify any industry related information concerning start-up and power escalation problems and did not focus on the priority of Reactor Power versus establishing Steam Generator control.

GP-005 was considered inadequate in that it did not contain prompts or action statements for reconciliation of discrepancies between the Intermediate Range Bi-stable Actuation and Power Range Nuclear Instrumentation indication.

Insufficient work practices also contributed to the event. Operators failed to use their independent indications, resulting in their inability to sense the core's actual power. When the Intermediate Range Bi-stable Actuation was received there was no comparison to the Power Range Nuclear Instrumentation, a form of error detection or verification of expected response.

Training content did not adequately address generic system information. The training received by the Control Room crew did not instill in the personnel the fundamental understanding of the inherent uncertainty of the Power Range Nuclear Instrumentation following a refueling of the reactor. Operations Management did not provide substantial direction as to the objectives of the Start-up training. The training content did not adequately address tools or equipment used to perform the task, potential consequences of inappropriate actions, and verification/self-checking practices. The training method inadequately presented the course materials.

Enclosure to Serial: RNP/93-3105

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), 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)	LER NUMBER (6)			PAGE (3)
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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

II. CAUSE OF EVENT (Continued)

The training which was developed as a result of the Reactivity Mismanagement event at CP&L's Harris Nuclear Plant was moved to Initial Training, and no continuing training was provided for the event. The training developed for the changes to the fuel cycle characteristics was developed by the Licensee's Corporate Fuels group with no augmentation by the site training department. The fuel cycle characteristic information was routed as Real Time Training Required Reading. The training material was not suitably configured for required reading and was not easily understood by Operators. Not all members of the crew performing the start-up had signed off on the required reading material prior to the start-up.

The start-up training requested by Operations management was limited to four hours. Some of the Control Room crew members were trained, during the start-up training, at different watch stations than they held during the start-up. The start-up training did not adequately cover the consequences of inappropriate actions nor did it address adequately the tools/equipment (RCS Delta-T indication) that would be used to perform the tasks.

III. ANALYSIS OF EVENT

This event had no adverse effect on plant safety as the core was maintained in a safe operating condition throughout the event. The plant did not exceed and would not have exceeded those limits important to safety for FSAR Chapter 15 events. The basis for Techncial Specification 3.10.7, "Power Ramp Rate Limits", states that the three percent limit is imposed to minimize the effects of adverse cladding stresses resulting from part power operation for extended periods of time.

This report is submitted pursuant to 10 CFR 50.73(a)(2)(i) as a condition prohibited by the plant's Technical Specifications.

NRC FORM 366A (5-92) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
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IV. CORRECTIVE ACTIONS

Adverse Condition Report 93-248 was initiated to determine the root cause of this event and to establish corrective actions. The schedule for corrective actions has been established commensurate with startup and safe operation activities. Prior to start-up, GP-005 will be revised to:

a) Use independent indications of Reactor Power to validate the Power Range NIs. Independent indicators will include: RCS Delta T, Intermediate Range NIs, First Stage Pressure and MWe. Instructions will assign responsibility for comparing the Power Range NIs with the independent indications.

b) Stop power escalation when any of the independent power indication parameters indicate that the reactor power has reached 20%, or if there is a significant discrepancy between the Reactor Power indicators when below 20% power.

c) At 20% Reactor power, verify that the indications of Reactor Power are within 5% of each other. (The 5% value is based on the value used in the Precautions and Limitations section of GP-005)

d) If the Power Ramp Rate Restrictions apply, revise GP-005 to:

1) Increase power to less than or equal to 30% at 3% per hour, based on the highest indication of Reactor Power.

2) Perform a Calorimetric and adjust the NIs, if applicable, prior to increasing power above 30%.

e) Stop power escalation when the independent power indication parameters show a deviation of Reactor power greater than 5% when the Reactor is above 20% power. (The 5% value is based on the value used in the Precautions and Limitations section of GP-005)

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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IV. CORRECTIVE ACTIONS (Continued)

Prior to each start-up after a refueling, simulator and/or classroom training will be conducted for operations crews, including STA and Reactor Engineers who are assigned to the shifts performing the start-up. Operations and Training management shall determine the crew training needs, and design the training to meet those needs. The following specific topics are will be considered.

a) The November 14,1993 HBR mismatch between actual and indicated reactor power.

b) New fuel cycle core changes and how the core, reactivity control and NIs may react differently from previous cycles.

c) Stressing the importance of monitoring diverse indications of Reactor Power, particularly for a new core start-up prior to the first calorimetric and adjustment of the NIs.

d) Appropriate industry events that have occurred during plant start-ups, such as those described in SOER 90-003.

e) Simulator scenarios should include realistic equipment malfunctions and NIS errors.

Training in the classroom and the simulator will be reviewed and enhanced to ensure that licensed operators have a fundamental knowledge of monitoring and controlling primary plant parameters. This training will consider application of this knowledge to changing plant conditions, including a review of applicable industry experience, and should be covered in retraining at a frequency that maintains the knowledge level.

V. ADDITIONAL INFORMATION

A. Component Failures

None

B. Previous Similar Events

None