NRC Form 366
(9-83)

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

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Catawba Nuclear Station, Unit 1	0 5 0 0 0 4 1 3	1 OF 0 4

TITLE (4)
Seventy Five Percent Power Exceeded During Testing

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 3	2 9	8 5	8 5	— 0 2 3 —	0 0	0 4	2 9	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 0 7 5

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)	[X] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 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	
	AREA CODE	
Roger W. Ouellette, Assistant Engineer, Licensing	7 0 4	3 7 3 - 7 5 3 0

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 fifteen single-space typewritten lines) (16)

On March 29, 1985, at 1625 hours, reactor power exceeded 75%. At the time, only two of three required test runs for Reactor Coolant (NC) System flow had been performed to meet the acceptance criteria specified in the Calimetric Reactor Coolant Flow Measurement Periodic Test. This test is conducted to comply with Technical Specification 4.2.3.2, which requires that NC System Flow be within certain limits.

The test was being performed in anticipation of increasing reactor power above 75% after the required power escalation testing at the 75% plateau was completed. Reactor power was being brought to approximately 74.5%, by deborating the NC System, to perform the required testing at this plateau. The Thermal Best Estimate, from the Operator Aid Computer was being observed since it is the most accurate indication of reactor power. However, at approximately 1600 hours, Thermal Best Estimate stopped trending upward and indicated an incorrect reactor power due to software problems. Due to periodic deboration, reactor power increased, but Thermal Best Estimate did not reflect this. Therefore, this incident is classified as a Design Deficiency.

When the software problem was recognized, reactor power was decreased, and by 2352 hours, was below 75%. This incident is reportable pursuant to 10 CFR 50.73 (a)(2)(i)(B).

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Technical Specification 4.2.3.2 requires that, prior to operation above 75% power, Reactor Coolant (NC) System flow and the ratio of measured to allowable Enthalpy Rise Hot Channel Factor be within certain limits. The NC System Flow requirement is met by Calimetric Reactor Coolant Flow Measurement Periodic Test, PT/1/A/4150/13B.

The Enthalpy Rise Hot Channel Factor is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power. The Enthalpy Rise Hot Channel Factor requirement is met by Core Power Distribution Periodic Test, PT/1/A/4150/05.

PT/1/A/4150/05 had been performed on March 27, 1985, at 68% power level, just two days prior to this incident. However, PT/1/A/4150/13B had not been completely performed prior to this incident. The test requires repeated runs of NC System flow measurement until the average total NC flow value of three runs is within $\pm$0.5% of each of the three runs comprising the average. Three NC flow runs were performed on March 27, 1985, with the second run not satisfying acceptance criteria. Therefore, a fourth run had to be performed on March 31, 1985. This run yielded acceptable results, but was performed after the increase to greater than 75% power.

The Thermal Output Calculations Program is used by plant personnel to perform necessary heat balance calculations around the NC System loops to determine loop flows, core power, and core burn-up, and to output these values as pseudo analog points, one of which is the "Thermal Best Estimate" of Reactor Power. Because of accuracy considerations, when secondary thermal power is less than 20%, the Thermal Best Estimate Program uses only inputs from the primary side. When secondary thermal power is more than 50%, the program uses only inputs from the secondary side. Between 20% and 50% power, the program utilizes both primary and secondary inputs by use of a weighting factor. Also, if one or more inputs from the primary (secondary) sides becomes invalid, the program reverts to the secondary (primary), to continue to provide a means of determining thermal power.

At 1600 hours, on March 29, 1985, the Power Escalation Shift Test Coordinator requested that the Nuclear Control Operator (NCO) increase Reactor Power to approximately 74.5% to perform 75% Plateau Testing. At this time, Thermal Best Estimate of Reactor Power indicated 73.2%. Also, at about the same time, the Power Range Nuclear Instrumentation (NI's) started slowly trending upward, while Thermal Best Estimate of reactor power remained the same. At 1618 hours and again at 1623 hours, the NCO injected non-borated water for about 1.5 minutes to increase reactor power. At 1625:14 hours, reactor power exceeded 75% as indicated by the NI's.

At 1700 hours, Personnel noticed a 2.5% mismatch in reactor power indication between the NI's and Thermal Best Estimate. The Shift Supervisor and NCO approached the Power Escalation Shift Test Coordinator about the problem, and asked which indication would be most reliable. The Test Coordinator stated that Thermal Best Estimate would be most accurate, and they should believe its indication. He also stated that the NI's would be calibrated at 75% power level to agree with Thermal Best Estimate.

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

Following the Test Coordinator's request for a reactor power increase at 1600 hours, the NCO had been adding load to the Turbine/Generator. After being told to believe Thermal Best Estimate at 1700 hours, which indicated 73.8% reactor power, the NCO resumed adding load to the Turbine/Generator (NI's indicated 76.3% reactor power at this point). Also, at 1755 hours, Feedwater Heater 1D1 was placed in service, which improved thermal cycle efficiency and allowed load to be added to the Turbine/Generator. From 1800 hours to 1850 hours, the NCO injected non-borated water 3 times to increase reactor power level.

During shift turnover at 1900 hours, the oncoming NCO was instructed by the NCO on duty to increase reactor power to 74.5% as indicated by Thermal Best Estimate (still indicating 73.8%). He was also informed of the problem of power indication mismatch. Immediately after shift turnover, the Test Coordinator entered the Control Room and reminded the NCO that 74.5% power was needed to perform the required testing. The NCO continued to add load to the Turbine/Generator, and from 1928 hours to 2059 hours, he injected non-borated water six times to increase reactor power level. During this time, Thermal Best Estimate remained at 73.8%, and the NI's increased to 81.9%. At 2100 hours, the NCO noticed an 8% mismatch between Thermal Best Estimate and the NI's, and brought the problem to the attention of the Test Coordinator. The Test Coordinator then obtained a Point Accumulation and Output (PAO) printout from the Operator Aid Computer (OAC) to observe the trend in Thermal Best Estimate versus NI's. He noticed a significant deviation since about 1600 hours, and that Thermal Best Estimate had not increased above 73.8%. He informed the NCO of a probable computer problem, and the NCO began inserting control rods to reduce reactor power level at 2124 hours. At 2125:14 hours, reactor power, as indicated by NI's, reached its maximum point during the incident, about 82.8%.

The Test Coordinator then contacted the Reactor Engineer about the problem, and was instructed to disable the secondary side input to the Thermal Best Estimate Program, making the program revert to using only primary side inputs. At 2258 hours, the secondary side inputs were disabled to the program, and the program then indicated that primary side Thermal Best Estimate exactly agreed with the NI's.

At 2352:45 hours, Reactor Power, as indicated by the NI's, had decreased to 74.9%. The Thermal Output Calculations Program was reassembled and implemented by 1830 hours on March 31, 1985.

At the same time that 75% Reactor Power was exceeded, the mismatch between Thermal Best Estimate and NI's was within $\pm 2\%$, as required by Technical Specification Table 4.3.1. Personnel were instructed to believe Thermal Best Estimate. At the time of the incident, no one had knowledge of the existing software problem. Also Delta-T indication of Reactor Power was not reliable at the time because it was initially calibrated conservatively, and recalibrated after determination of full power Delta-T. Therefore, this incident is classified as a Design Deficiency, due to the software problems. Subsequent investigation into the software problem attributed the calculation error to the magnitude limits of single precision floating point arithmetic, i.e., the large numbers used in secondary side calculations were being truncated and significant digits

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

lost. The Thermal Output Calculations Program was modified, reassembled, and reimplemented using double precision by 1830 hours on March 31, 1985.

The resulting thermal cycle efficiency improvement from placing Feedwater Heater 1D1 in service at 1755 hours allowed an addition of secondary load without a necessary load addition to the primary side. This, along with the problem mentioned above, led the NCO to believe that the reactor was actually below 75% power. However, the increasing mismatch between Thermal Best Estimate and NI's should have been more rapidly identified.

CORRECTIVE ACTION

1) The NCO began reducing Reactor Power after identification of the software problem, reducing power level to below 75%.

2) The Thermal Output Calculations Program was modified, reassembled, and reimplemented utilizing double precision by the Process Computer Group.

3) This incident was discussed with all licensed Operations Personnel.

SAFETY ANALYSIS

The final NC flow run was performed on March 31, 1985, yielding satisfactory results. It is believed that a secondary side transient caused changing plant thermal conditions and consequently erroneous data for the third NC flow run, making a fourth run necessary. Therefore, although the final NC flow run had not been conducted prior to occurrence of this incident, acceptable values for NC flow can be assumed during this incident.

An acceptable value of the ratio of measured to allowable Enthalpy Rise Hot Channel Factor was obtained during core physics testing performed at 68% Reactor Power on March 27, 1985, just two days prior to this incident. This testing met the surveillance requirements in Technical Specification 4.2.3.2b.

Acceptable Testing results for Reactor Coolant flow rate and Enthalpy Rise Hot Channel Factor ensure that, 1) The design limits on peak local power density and minimum Departure from Nucleate Boiling Ratio are not exceeded, and 2) In the event of a LOCA, the peak fuel clad temperature will not exceed the 2200°F Emergency Core Cooling System Acceptance Criteria Limit. Therefore, the Reactor was in a safe operating condition throughout this incident.

Futhermore, the Reactor Trip Setpoint for the power range NI's during this incident was 95%. An even more conservative reactor trip was the overpower Delta-T Setpoint, which was 91%. However, before Reactor Trip would have occurred, a block of automatic and manual rod withdrawal and Turbine runback would have occurred at 88%. Therefore, unintentional power escalation would not have gone unchecked.

The health and safety of the public were not affected by this incident.

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HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

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April 29, 1985

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Catawba Nuclear Station, Unit 1
Docket No. 50-413

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Licensee Event Report 413/85-23 concerning reactor power being increased above 75% RTP during 75% power testing. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

H.B. Tucker /RWO

Hal B. Tucker

RWO:slb

Attachment

cc: Dr. J. Nelson Grace, Regional Administrator
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