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): Catawba Nuclear Station, Unit 2

DOCKET NUMBER (2): 0 5 0 0 0 4 1 4

PAGE (3): 1 OF 0 4

TITLE (4): Feedwater Isolation While Cycling Steam Generator Power Operated Relief Valve Due To Unknown Cause

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 4	2 7	8 8	8 8	— 0 1 8 —	0 0	0 5	2 7	8 8	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 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)	[X] 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)	[ ] 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: Julio G. Torre, Associate Engineer - Licensing

TELEPHONE NUMBER: AREA CODE 7 0 4 | 3 7 3 - 8 0 2 9

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)

[ ] 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 April 27, 1988, at 1625 hours, an unexpected Steam Generator (S/G) 2C Hi Hi Level Turbine Trip signal occurred while cycling the S/G 2C Power Operated Relief Valve (PORV). The Hi Hi Level signal occurred immediately when the Control Room Operator began opening the PORV for S/G 2C. A Feedwater Isolation was initiated upon receipt of 2 out of 4 channels indicating Hi Hi Level. Plant response was minimal as the Unit was in Mode 5, Cold Shutdown, at the time. Control Room Operators subsequently realigned affected valves and reset the Feedwater Isolation.

This investigation revealed no reason for the spurious Hi Hi Level signal for S/G 2C. Previous incidents have identified instability of S/G Narrow Range level indication in Mode 4, Hot Shutdown, and Mode 5, Cold Shutdown as the probable cause, when rapid S/G pressure changes have occurred as a result of valve motion.

The health and safety of the public were unaffected by this event.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Catawba Nuclear Station, Unit 2	0 5 0 0 0 4 1 4	8 8	— 0 1 8	— 0 0	0 2 OF 0 4

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

BACKGROUND:

The Steam Generator (EIIS:SG) (S/G) Power Operated Relief Valves (EIIS:V) (PORVs) are provided for overpressure protection of the Main Steam (EIIS:SB) Lines and to minimize operation of the safety valves. OP/2/A/6100/02, Controlling Procedure for Unit Shutdown, requires cycling of the S/G PORVs to verify depressurization of the S/Gs.

Each S/G is provided with four channels of narrow range level indication. If 2 out of 4 S/G channels exceed 78% on Unit 2, a Hi Hi Level signal (P-14) is initiated causing an automatic Turbine Trip and Feedwater Isolation.

DESCRIPTION OF INCIDENT:

On April 24, 1988, Operations personnel began Unit 2 shutdown per OP/2/A/6100/02, Controlling Procedure for Unit Shutdown. At approximately 1300 hours on April 27, 1988, Unit 2 entered Mode 5, Cold Shutdown.

At 1401 hours, Control Room Operators (CROs) had isolated the Main Steam Isolation Valves (MSIVs) and their associated bypasses (MSBIVs). The S/Gs were secured from reverse purge at 1402 hours. S/Gs 2A , 2B, and 2D were aligned to the Nitrogen supply at 1450 hours. S/G 2C was not aligned to the Nitrogen supply due to work being performed on 2SA4, Main Steam 2C to Auxiliary Feedwater (EIIS:BA) Pump (EIIS:P) No. 2 Maintenance Isolation Valve. The CRO verified that Reactor Coolant (EIIS:AB) and core exit temperatures were less than 160 degrees F and less than 200 degrees F respectively, at 1606 hours.

At approximately 1610 hours, the CRO verified S/G pressures to be zero psig. This was verified by utilizing Operator Aid Computer points and S/G pressure gauges. The CRO made these verifications prior to beginning cycling of PORVs as required by the Shutdown Procedure. Cycling of these valves ensures that zero steam pressure is present prior to removing the final Reactor Coolant Pump from service during shutdown. This procedure also allows verifying S/G pressure by using a vent or by verifying steamline temperatures to be less than 200 degrees F. The CRO stated that the steamline temperatures were greater than 200 degrees F.

The CRO commenced cycling of S/G PORVs and successfully verified S/G 2A, 2B, and 2D to be depressurized without incident. At 1623:51 hours, the CRO began opening S/G 2C PORV. S/G 2C Channel 1 Hi Hi Level signal occurred at 1624:10:293 hours, then S/G 2C Channel 2 Hi Hi Level signal occurred at 1624:11:683 hours. The combination of 2 of 4 Hi Hi Level channels satisfied logic for initiation of a Feedwater Isolation. 2CF48, S/G 2C Feedwater Control Bypass valve, and associated CF to Auxiliary Feedwater (CA) bypass valves automatically closed as designed. The S/G 2C Channel 2 Hi Hi Level signal cleared at 1624:14:307 hours. At 1624:17:253 hours, the S/G 2C Channel 1 Hi Hi Level signal cleared.

The CROs reset the Feedwater Isolation signal and realigned appropriate valves. The procedure step for cycling of the S/G PORVs was signed off at 1635 hours. The Shift Supervisor notified the Nuclear Regulatory Commission of the Feedwater Isolation at 1811 hours.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Catawba Nuclear Station, Unit 2	0 5 0 0 0 4 1 4	8 8	— 0 1 8	— 0 0	0 3 OF 0 4

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

CONCLUSION:

No evidence exists to determine the cause of the spurious Hi Hi Level signal in S/G 2C. The narrow range level chart recorder indicates a spike from approximately 50% to 79%. The S/G pressure chart recorder indicated constant pressure of zero psig during the spurious signal which should have eliminated level change due to differential pressure. S/G 2C was not aligned to the Nitrogen supply as were the other three S/Gs. Verification that this alignment could have caused the spurious level spike is inconclusive. The spurious signal cleared in approximately 6 seconds. Control Room Operators then reset the Feedwater Isolation.

It has been determined during previous testing that the instrumentation used by the CROs to indicate steam line pressure cannot indicate negative pressures. Since S/G 2C was not pressurized with Nitrogen, it is possible that a vacuum existed in the S/G at the time the PORV was opened. This would have caused an abrupt pressure change which has caused the level instrumentation to spike in the past. Duke Power Instrumentation and Electrical (IAE) has committed to perform testing on S/G 2C by the End of Cycle 2 Refueling Outage to attempt to determine the specific cause of spurious ESF actuations when the S/Gs are subjected to pressure pulses. S/G 2C level instrumentation has been shown to be the most sensitive to pressure pulses in the past. This report will be revised if the specific cause of the spurious ESF actuation can be determined.

Operations, Westinghouse, and Integrated Scheduling are pursuing a proposed modification to relocate Unit 2 S/G Narrow Range Level taps which will provide a less stringent level span. Tentatively, the proposed modification is expected to be implemented during the next Unit 2 refueling outage. The decision to implement the proposed modification was based on data now being collected from Nuclear Station Modification (NSM) 20303, which was installed on March 16, 1988. This NSM installed additional instrumentation on S/G 2C so data could be gathered to support lowering the narrow range tap.

There has been one previous incident of Feedwater Isolation due to S/G level exceeding P-14 due to Unknown cause (see LER 414/86-01). Design Deficiencies were attributed to three more incidents of Feedwater Isolation: LER 413/87-47 (No precision instrumentation for indication of low pressures), LER 414/86-03 (instrumentation calibration for Unit 2 S/Gs inadequate), and LER 414/86-26 (Design of Main Steam Isolation Bypass Valves do not allow for complete equalization across Main Steam Isolation Valves). In addition, LER 414/87-14 was written, Cause Code Unknown, after Feedwater Isolation on S/G 2A Lo Lo level, and LER 413/86-15, Defective Procedure, was written due to Feedwater Isolation caused by S/G Hi Hi Level. All of these incidents involved valve manipulation which caused abrupt pressure changes on the S/Gs. These incidents occurred in Mode 4, Hot Shutdown, and Mode 5, Cold Shutdown.

CORRECTIVE ACTIONS:

SUBSEQUENT

(1) Operations personnel reset Feedwater Isolation.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Catawba Nuclear Station, Unit 2	0 5 0 0 0 4 1 4	8 8	— 0 1 8	— 0 0	0 4 OF 0 4

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

(2) Operations personnel returned CF/CA to original alignment.

PLANNED

(1) Station personnel will perform testing on S/G 2C to attempt to determine the specific cause of spurious ESF actuations which occur when the S/Gs are subjected to abrupt pressure changes in Mode 4 and Mode 5.

(2) This report will be revised if the specific cause is determined for spurious P-14 actuations .

(3) A decision will be reached with regard to the proposed NSM to relocate Unit 2 S/G narrow range level taps.

SAFETY ANALYSIS:

During receipt of the Steam Generator (S/G) Hi Hi Level (P-14) signal, all appropriate CF and CA bypass control valves actuated. S/G narrow range levels remained at approximately 50% and were not affected by the Feedwater Isolation. Decay heat was being removed by the Residual Heat Removal System, which was not affected by the P-14 signal. Therefore, an adequate heat sink was available at all times.

This incident is reportable pursuant to 10 CFR 50.73, Section (a)(2)(iv).

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

DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

May 27, 1988

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

Subject: Catawba Nuclear Station, Unit 2
Docket No. 50-414
LER 414/88-18

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Licensee Event Report 414/88-18 concerning a feedwater isolation while cycling Steam Generator Power Operated Relief Valve. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Hal B. Tucker /mn

Hal B. Tucker

JGT/29/sbn

Attachment

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Mr. P. K. Van Doorn
NRC Resident Inspector
Catawba Nuclear Station

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Group File: CN-815.04

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): Catawba Nuclear Station, Unit 2

DOCKET NUMBER (2): 0 5 0 0 0 4 1 4

PAGE (3): 1 OF 0 4

TITLE (4): Feedwater Isolation While Cycling Steam Generator Power Operated Relief Valve Due To Unknown Cause

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 4	2 7	8 8	8 8	0 1 8	0 0	0 5	2 7	8 8	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 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)	X 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)	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	
Julio G. Torre, Associate Engineer - Licensing	7 0 4	3 7 3 - 8 0 2 9

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)

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 April 27, 1988, at 1625 hours, an unexpected Steam Generator (S/G) 2C Hi Hi Level Turbine Trip signal occurred while cycling the S/G 2C Power Operated Relief Valve (PORV). The Hi Hi Level signal occurred immediately when the Control Room Operator began opening the PORV for S/G 2C. A Feedwater Isolation was initiated upon receipt of 2 out of 4 channels indicating Hi Hi Level. Plant response was minimal as the Unit was in Mode 5, Cold Shutdown, at the time. Control Room Operators subsequently realigned affected valves and reset the Feedwater Isolation.

This investigation revealed no reason for the spurious Hi Hi Level signal for S/G 2C. Previous incidents have identified instability of S/G Narrow Range level indication in Mode 4, Hot Shutdown, and Mode 5, Cold Shutdown as the probable cause, when rapid S/G pressure changes have occurred as a result of valve motion.

The health and safety of the public were unaffected by this event.

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* U.S.GPO:1984-0-464-481/18750

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Catawba Nuclear Station, Unit 2	0 5 0 0 0 4 1 4	8 8	— 0 1 8	— 0 0	0 2 OF 0 4

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

BACKGROUND:

The Steam Generator (EIIS:SG) (S/G) Power Operated Relief Valves (EIIS:V) (PORVs) are provided for overpressure protection of the Main Steam (EIIS:SB) Lines and to minimize operation of the safety valves. OP/2/A/6100/02, Controlling Procedure for Unit Shutdown, requires cycling of the S/G PORVs to verify depressurization of the S/Gs.

Each S/G is provided with four channels of narrow range level indication. If 2 out of 4 S/G channels exceed 78% on Unit 2, a Hi Hi Level signal (P-14) is initiated causing an automatic Turbine Trip and Feedwater Isolation.

DESCRIPTION OF INCIDENT:

On April 24, 1988, Operations personnel began Unit 2 shutdown per OP/2/A/6100/02, Controlling Procedure for Unit Shutdown. At approximately 1300 hours on April 27, 1988, Unit 2 entered Mode 5, Cold Shutdown.

At 1401 hours, Control Room Operators (CROs) had isolated the Main Steam Isolation Valves (MSIVs) and their associated bypasses (MSBIVs). The S/Gs were secured from reverse purge at 1402 hours. S/Gs 2A , 2B, and 2D were aligned to the Nitrogen supply at 1450 hours. S/G 2C was not aligned to the Nitrogen supply due to work being performed on 2SA4, Main Steam 2C to Auxiliary Feedwater (EIIS:BA) Pump (EIIS:P) No. 2 Maintenance Isolation Valve. The CRO verified that Reactor Coolant (EIIS:AB) and core exit temperatures were less than 160 degrees F and less than 200 degrees F respectively, at 1606 hours.

At approximately 1610 hours, the CRO verified S/G pressures to be zero psig. This was verified by utilizing Operator Aid Computer points and S/G pressure gauges. The CRO made these verifications prior to beginning cycling of PORVs as required by the Shutdown Procedure. Cycling of these valves ensures that zero steam pressure is present prior to removing the final Reactor Coolant Pump from service during shutdown. This procedure also allows verifying S/G pressure by using a vent or by verifying steamline temperatures to be less than 200 degrees F. The CRO stated that the steamline temperatures were greater than 200 degrees F.

The CRO commenced cycling of S/G PORVs and successfully verified S/G 2A, 2B, and 2D to be depressurized without incident. At 1623:51 hours, the CRO began opening S/G 2C PORV. S/G 2C Channel 1 Hi Hi Level signal occurred at 1624:10:293 hours, then S/G 2C Channel 2 Hi Hi Level signal occurred at 1624:11:683 hours. The combination of 2 of 4 Hi Hi Level channels satisfied logic for initiation of a Feedwater Isolation. 2CF48, S/G 2C Feedwater Control Bypass valve, and associated CF to Auxiliary Feedwater (CA) bypass valves automatically closed as designed. The S/G 2C Channel 2 Hi Hi Level signal cleared at 1624:14:307 hours. At 1624:17:253 hours, the S/G 2C Channel 1 Hi Hi Level signal cleared.

The CROs reset the Feedwater Isolation signal and realigned appropriate valves. The procedure step for cycling of the S/G PORVs was signed off at 1635 hours. The Shift Supervisor notified the Nuclear Regulatory Commission of the Feedwater Isolation at 1811 hours.

CONCLUSION:

No evidence exists to determine the cause of the spurious Hi Hi Level signal in S/G 2C. The narrow range level chart recorder indicates a spike from approximately 50% to 79%. The S/G pressure chart recorder indicated constant pressure of zero psig during the spurious signal which should have eliminated level change due to differential pressure. S/G 2C was not aligned to the Nitrogen supply as were the other three S/Gs. Verification that this alignment could have caused the spurious level spike is inconclusive. The spurious signal cleared in approximately 6 seconds. Control Room Operators then reset the Feedwater Isolation.

It has been determined during previous testing that the instrumentation used by the CROs to indicate steam line pressure cannot indicate negative pressures. Since S/G 2C was not pressurized with Nitrogen, it is possible that a vacuum existed in the S/G at the time the PORV was opened. This would have caused an abrupt pressure change which has caused the level instrumentation to spike in the past. Duke Power Instrumentation and Electrical (IAE) has committed to perform testing on S/G 2C by the End of Cycle 2 Refueling Outage to attempt to determine the specific cause of spurious ESF actuations when the S/Gs are subjected to pressure pulses. S/G 2C level instrumentation has been shown to be the most sensitive to pressure pulses in the past. This report will be revised if the specific cause of the spurious ESF actuation can be determined.

Operations, Westinghouse, and Integrated Scheduling are pursuing a proposed modification to relocate Unit 2 S/G Narrow Range Level taps which will provide a less stringent level span. Tentatively, the proposed modification is expected to be implemented during the next Unit 2 refueling outage. The decision to implement the proposed modification was based on data now being collected from Nuclear Station Modification (NSM) 20303, which was installed on March 16, 1988. This NSM installed additional instrumentation on S/G 2C so data could be gathered to support lowering the narrow range tap.

There has been one previous incident of Feedwater Isolation due to S/G level exceeding P-14 due to Unknown cause (see LER 414/86-01). Design Deficiencies were attributed to three more incidents of Feedwater Isolation: LER 413/87-47 (No precision instrumentation for indication of low pressures), LER 414/86-03 (instrumentation calibration for Unit 2 S/Gs inadequate), and LER 414/86-26 (Design of Main Steam Isolation Bypass Valves do not allow for complete equalization across Main Steam Isolation Valves). In addition, LER 414/87-14 was written, Cause Code Unknown, after Feedwater Isolation on S/G 2A Lo Lo level, and LER 413/86-15, Defective Procedure, was written due to Feedwater Isolation caused by S/G Hi Hi Level. All of these incidents involved valve manipulation which caused abrupt pressure changes on the S/Gs. These incidents occurred in Mode 4, Hot Shutdown, and Mode 5, Cold Shutdown.

CORRECTIVE ACTIONS:

SUBSEQUENT

(1) Operations personnel reset Feedwater Isolation.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

(2) Operations personnel returned CF/CA to original alignment.

PLANNED

(1) Station personnel will perform testing on S/G 2C to attempt to determine the specific cause of spurious ESF actuations which occur when the S/Gs are subjected to abrupt pressure changes in Mode 4 and Mode 5.

(2) This report will be revised if the specific cause is determined for spurious P-14 actuations .

(3) A decision will be reached with regard to the proposed NSM to relocate Unit 2 S/G narrow range level taps.

SAFETY ANALYSIS:

During receipt of the Steam Generator (S/G) Hi Hi Level (P-14) signal, all appropriate CF and CA bypass control valves actuated. S/G narrow range levels remained at approximately 50% and were not affected by the Feedwater Isolation. Decay heat was being removed by the Residual Heat Removal System, which was not affected by the P-14 signal. Therefore, an adequate heat sink was available at all times.

This incident is reportable pursuant to 10 CFR 50.73, Section (a)(2)(iv).

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

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(704) 373-4531

May 27, 1988

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

Subject: Catawba Nuclear Station, Unit 2
Docket No. 50-414
LER 414/88-18

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Licensee Event Report 414/88-18 concerning a feedwater isolation while cycling Steam Generator Power Operated Relief Valve. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Hal B. Tucker/mn

Hal B. Tucker

JGT/29/sbn

Attachment

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NRC Resident Inspector
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