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)
Millstone Nuclear Power Station Unit 3	0 5 0 0 0 4 2 3	1 OF 0 2

TITLE (4)

Feedwater Isolation on Opening the "A" Main Steam Isolation Valve

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 2	1 4	8 6	8 6	0 1 8	0 0	0 3	1 4	8 6		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	3	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more of the following) (11)			
POWER LEVEL (10)	0 0 0	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)	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
Jeffrey A. Langan, Associate Engineer	AREA CODE 2 0 3 4 4 7 - 1 7 9 1

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	
D	S J	I S V	S 4 5 0	N							

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 2/14/86, at 1008 hours, while operating at 0% power in the Hot Standby mode, the plant experienced a feedwater isolation due to high-high level in Steam Generator A. The feedwater isolation resulted when opening the Main Steam Isolation Valve (MSIV) on Steam Generator A caused a swell in the generator.

Prior to opening the A MSIV, Operations personnel pressurized the main steam header downstream of the MSIV using the Main Steam Isolation Bypass valve, as required by procedure. When the differential pressure across the isolation valve was 40 psid, as indicated by main board instruments and the plant process computer, the A Main Steam Isolation Valve was opened. This caused level in the A Steam Generator to swell to 92% of narrow range level. A feedwater isolation occurred when narrow range level reached the high-high setpoint of 80%.

To prevent this from reoccurring, the operating procedure has been revised to include a caution warning of possible feedwater isolation when opening an MSIV, and specifying several courses of action operators may take to prevent it.

This report is being submitted in accordance with 10CFR50.73 (a) (2) (iv).

NRC Form 366 (9-83)

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

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

On 2/14/86, at 1008 hours, while operating at 0% power in the Hot Standby mode, the plant experienced a feedwater isolation due to high-high level in Steam Generator A. The feedwater isolation resulted when opening the Main Steam Isolation Valve (MSIV) on Steam Generator A caused a swell in the generator.

Prior to opening the A MSIV, Operations personnel pressurized the main steam header downstream of the MSIV using the Main Steam Isolation Bypass Valve, as required by procedure. When the differential pressure across the isolation valve was 40 psid, as indicated by main board instruments and the plant process computer, the A Main Steam Isolation Valve was opened. This caused level in the A Steam Generator to swell to 92% of narrow range level. A feedwater isolation occurred when narrow range level reacned the high-high setpoint of 80%. The feedwater isolation also resulted in the motor driven feedwater pump tripping off line.

The high-high steam generator level condition rapidly cleared, and was manually reset. Within 4 minutes the motor driven feedwater pump was restarted, and the remaining three MSIV's opened. This is the first feedwater isolation that has occurred as a result of swelling in a steam generator caused by opening a MSIV. To prevent this from reoccurring, the operating procedure has been revised to include a caution warning of a possible feedwater isolation when opening an MSIV, and specifying several courses of action operators may take to prevent it.

This report is being submitted in accordance with 10CFR50.73 (a) (2) (iv).

File

NORTHEAST UTILITIES
THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

General Offices • Selden Street, Berlin, Connecticut

P.O. BOX 270
HARTFORD, CONNECTICUT 06141-0270
(203) 666-6911

June 16, 1986
MP-9180

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

Reference: Facility Operating License No. NPF-49
Docket No. 50-423
Licensee Event Report 50-423/86-036-00

Gentlemen:

This letter forwards Licensee Event Report 86-036-00 required to be submitted within thirty days pursuant to 10CFR50.73 (a) (2) (i) (B), any operation or condition prohibited by the Plant's Technical Specifications.

Yours truly,

NORTHEAST NUCLEAR ENERGY COMPANY

Wayne D. Romberg
Station Superintendent
Millstone Nuclear Power Station

WDR/BMP:se

Attachment: LER 86-036-00

cc: Dr. T. E. Murley, Region I

FOIA-87-512

B/5

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3pp.

NRC Form 366

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)
Millstone Nuclear Power Station Unit 3	0 5 0 0 0 4 2 3	1 OF 0 3

TITLE (4)

Violation of Plant Technical Specifications - Non-Compliance with Action Statement

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 9	8 6	8 6	0 3 6	0 0	0 6	1 6	8 6		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	1 0 0	20.402(b)	20.405(c)	50.73(a)(2)(iii)	73.71(b)
		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(iv)	73.71(c)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(v)	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)(vii)	
		20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(A)	
		20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(viii)(B)	
				50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Barry Pinkowitz, Senior Engineer	AREA CODE 2 0 3 4 4 7 - 1 1 7 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	
B	E J			N							

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 5/19/86 at 1415 hours while operating at 100% power, it was discovered that the plant had been operating in the Action Statement of Plant Technical Specification 3.8.2.1.b in that battery bank 301A-2 was not operable due to an unperformed modification to battery charger 301A-2 to correct a condition which could result in equipment impact due to a seismic event.

Within 24 hours of discovery, seismic bracing was designed and installed on battery charger 301A-2 restoring the equipment to operability.

This report is being submitted in accordance with 10CFR50.73 (a) (2) (i) (B).

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

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

On 5/19/86 at 1415 hours while operating at 100% power, it was discovered that the plant had been operating in the Action Statement of Plant Technical Specification 3.8.2.1.b in that battery bank 301A-2 was not operable due to an unperformed modification to battery charger 301A-2 to correct a potential seismic interaction between charger 301A-2 and inverter INV-3. Affected battery bank 301A-2 was immediately declared inoperable. Within the 24 hours allowed by Technical Specification 3.8.2.1.b seismic bracing was designed and installed on battery bank 301A-2 restoring the equipment to operability. The three remaining safety related battery banks had their modifications complete. For the long term, the permanent modification will be performed at the next convenient outage and the bracing removed.

There were no safety implications to the public as only one channel of a four independent channel system was inoperable. The failure of this one power supply would not prevent the reactor protection system or engineered safety feature system from performing their functions nor would it prevent the safe shutdown of the plant.

This report is being submitted in accordance with 10CFR50.73 (a) (2) (i) (B).

PACIFIC GAS AND ELECTRIC COMPANY

PG&E — 77 BEALE STREET • SAN FRANCISCO, CALIFORNIA 94106 • (415) 781-4211 • TWX 910-372-6587

JAMES D. SHIFFER
VICE PRESIDENT
NUCLEAR POWER GENERATION

May 4, 1987

PGandE Letter No.: DCL-87-098

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

Re: Docket No. 50-275, OL-DPR-80
Diablo Canyon Unit 1
Licensee Event Report 1-86-017-00
Tipped Fuel Assembly During Refueling

Gentlemen:

PGandE is submitting the enclosed voluntary Licensee Event Report concerning the tipping of a fuel assembly during refueling. This report is being submitted for information purposes only as described in Item 19 of Supplement Number 1 to NUREG-1022.

This event has in no way affected the public's health and safety.

Kindly acknowledge receipt of this material on the enclosed copy of this letter and return it in the enclosed addressed envelope.

Sincerely,

Enclosure

cc: L. J. Chandler
J. B. Martin
M. M. Mendonca
P. P. Narbut
B. Norton
C. M. Trammell
CPUC
Diablo Distribution
INPO

DC1-86-TN-N124
1243S/0050K/RHM/1271

FOIA-87-512
B/6

8705110140
5pp.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
DIABLO CANYON UNIT 1	0 5 0 0 0 2 7 5	1 OF 0 4

TITLE (4)
TIPPED FUEL ASSEMBLY DURING REFUELING

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)		
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR
1 0	0 1	8 6	8 6	0 1 1 7	0 0	0 5	0 4	8 7

OTHER FACILITIES INVOLVED (8)	
FACILITY NAMES	DOCKET NUMBER(S)
	0 5 0 0 0
	0 5 0 0 0

OPERATING MODE (9)	6
POWER LEVEL (10)	0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (11)

- [ ] 10 CFR _____
- [X] OTHER (Specify in Abstract below and in Text, NRC Form 366A)

Voluntary Report

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
RICHARD M. LUCKETT, REGULATORY COMPLIANCE ENGINEER	8 0 5 5 4 5 - 7 8 5 1

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 (16)

This voluntary LER is being submitted for information purposes only as described in Item 19 of Supplement Number 1 to NUREG-1022.

On October 1, 1986, while Unit 1 was in Mode 6 (Refueling), fuel assembly A-06 tipped against the core baffle, damaging the corner of the assembly grid strap. The event occurred after the fuel assembly had been raised slightly above the core support plate to allow the resetting of the load limit switch on the fuel manipulator crane. A problem developed with resetting the load limit switch, and the fuel assembly was set back down. Subsequently the shift ended, and oncoming shift personnel discovered that the fuel assembly had tipped from its normal upright position and was leaning against the core baffle plates.

The fuel assembly was righted and removed from the core in accordance with an approved procedure. With the exception of a corner on grid strap Number 4, no damage to the fuel assembly was identified.

To prevent recurrence, the core unloading sequence procedure (OP B-8DS1) was revised to emphasize the need for positive indication when reseating a fuel assembly.

This event had no safety consequences and in no way affected the health and safety of the public.

1243S/0050K

I. Initial Conditions

Unit 1 was in Mode 6 (Refueling), at a temperature of 76°F and at atmospheric pressure.

II. Description of Event

A. Event:

On October 1, 1986, while Unit 1 was in Mode 6 (Refueling), fuel assembly A-06 tipped against the core baffle plate, damaging the corner of the assembly (AC) grid strap. The event occurred after the fuel assembly was raised slightly to allow the resetting of the load limit switch (WIS) on the fuel manipulator crane (DF)(FHM). This resetting of the load limit switch is required by procedure prior to movement of the first fuel assembly containing a control rod (AA). A problem developed with resetting the load limit switch, and the process was halted. The fuel assembly was set back down and unlatched until the problem could be resolved. Subsequently the shift ended, and the oncoming shift discovered that the fuel assembly had tipped from its normal upright position sometime during the two-hour shift change period and was leaning against the core baffle plates. Further investigation determined that when the fuel assembly was set back down, it was not aligned with the index pins in the lower core support plate due to fuel rod bowing. The visual inspection prior to unlatching the assembly failed to detect this condition due to a combination of lighting conditions and the glow from the Cerenkov radiation.

The fuel assembly was righted and removed from the core in accordance with temporary procedure TP TO-8610. No radiation release was observed by airborne monitors or from sampling of the refueling cavity water. A followup inspection identified damage as a torn corner on assembly grid strap Number 4. No fuel rod damage occurred to assembly A-06 or to any surrounding assemblies in the core. Assembly A-06 was placed in the spent fuel pool for storage. This assembly was not used in the second cycle, and therefore no repair to the grid strap or corrective action was necessary.

B. Inoperable structures, components, or systems that contributed to the event:

None.

C. Dates and approximate times for major occurrences:

1. October 1, 1986, 0800 PDT: Event date

2. October 1, 1986, 1000 PDT: Discovery date

3. October 3, 1986, 0356 PDT: Fuel assembly righted and placed in spent fuel pool

D. Other systems or secondary functions affected:

None

E. Method of discovery:

During inspection after shift change.

F. Operator actions:

None required

G. Safety system response:

None required

III. Cause of Event

A. Immediate cause:

Failure to reseat fuel assembly A-06 on lower core support plate index pins.

B. Root cause:

A combination of bowed fuel rods and lighting conditions contributed to the operators' failure to detect that the assembly was not aligned with the index pins.

IV. Analysis of Event

Tipping of the fuel assembly against the core baffle did not cause any damage to fuel rods nor compromise the integrity of the fission product barrier provided by the fuel cladding. Grid strap damage during fuel evolutions is a common industry occurrence and presents no safety consequences unless the assembly is reused, at which time the grid strap damage would be evaluated to ensure there is no potential for causing fuel rod damage. Thus, no adverse consequences or implications resulted from this event.

V. Corrective Actions

To prevent recurrence, the core unloading sequence procedure, OP B-8DS1, was revised to emphasize the need for positive indication when reseating a fuel assembly.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
DIABLO CANYON UNIT 1	0 5 0 0 0 2 7 5	8 6	— 0 1 7	— 0 0	0 4 OF 0 4

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

VI. Additional Information

A. Failed components:

None

B. Previous LERs on similar events:

None

1243S/0050K