



Entergy Nuclear Generation Company  
Pilgrim Nuclear Power Station  
600 Rocky Hill Road  
Plymouth, MA 02360

**J. F. Alexander**  
Director  
Nuclear Assessment

October 25, 1999  
ENGCLtr. 2.99.112

10 CFR 50.73

U.S. Nuclear Regulatory Commission  
ATTN: Document Control Desk  
Washington, DC 20555

Docket No. 50-293  
License No. DPR-35

The enclosed Licensee Event Report (LER) 99-011-00, "Postulated Fire in Cable Spreading Room Potentially Affecting Safe Shutdown," is submitted in accordance with 10 CFR 50.73.

Except for the submittal of a supplement to this report, this letter contains no commitments.

Please do not hesitate to contact me if there are any questions regarding this report.

Sincerely,

A handwritten signature in black ink, appearing to read "J.F. Alexander".

J.F. Alexander

DWE/sc  
Enclosure

cc: Mr. Hubert J. Miller  
Regional Administrator, Region 1  
U.S. Nuclear Regulatory Commission  
475 Allendale Road  
King of Prussia, PA 19406

INPO Records  
700 Galleria Parkway  
Atlanta, GA 30339-5957

Sr. NRC Resident Inspector  
Pilgrim Nuclear Power Station

IE22

# LICENSEE EVENT REPORT (LER)

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

Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), 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. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

FACILITY NAME (1)

PILGRIM NUCLEAR POWER STATION

DOCKET NUMBER (2)

05000-293

PAGE(3)

1 of 1

TITLE (4)

Postulated Fire in Cable Spreading Room Potentially Affecting Safe Shutdown

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
09	22	1999	1999	011	00	10	25	99	N/A	05000
									N/A	05000

  

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (Check one or more) (11)			
POWER LEVEL (10)	100	20.2201 (b)		20.2203(a)(2)(v)	
		22.2203(a)(1)		20.2203(a)(3)(i)	X
		20.2203(a)(2)(i)		20.2203(a)(3)(ii)	
		20.2203(a)(2)(ii)		20.2203(a)(4)	
		20.2203(a)(2)(iii)		50.36(c)(1)	
		20.2203(a)(2)(iv)		50.36(c)(2)	

LICENSEE CONTACT FOR THIS LER (12)

NAME

Douglas W. Ellis - Regulatory Affairs Senior Engineer

TELEPHONE NUMBER (Include Area Code)  
(508) 830-8160

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)

<input checked="" type="checkbox"/>	YES					EXPECTED SUBMISSION DATE(15)	MONTH	DAY	YEAR
(If yes, complete EXPECTED SUBMISSION DATE)							11	15	99

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On September 22, 1999, a condition outside the design basis was identified. Specifically, a postulated fire in the cable spreading room (CSR) could potentially affect the alternate shutdown panel of an emergency diesel generator (EDG) and hence, the ability of one (or both) EDG(s) to sufficiently power AC powered electrical loads necessary to achieve or maintain safe shutdown. The postulated fire could limit EDG electrical output because EDG watt-meter cabling in the CSR was not protected as part of 10 CFR 50 Appendix R modifications.

The preliminary root cause analysis could not determine the exact cause. The original design of the alternate shutdown panels was developed in the 1978 - 1979 time frame and additional modifications were added in the 1986 - 1987 time frame. Fire watches for the potentially affected alternate shutdown panels were established in accordance with Technical Specification 3.12. The operability of the EDGs is not affected. This report will be supplemented after the root cause analysis is completed. Corrective action being evaluated includes revision of a procedure for a shutdown outside of the control room.

The condition was identified during power operation while at 100 percent reactor power. The reactor vessel pressure was about 1034 psig with the reactor vessel water temperature at the saturation temperature for that pressure. The condition posed and poses no threat to public health and safety.