VERMONT YANKEE NUCLEAR POWER CORPORATION

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BVY 96-21

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

REFERENCE: Operating License DPR-28
Docket No. 50-271
Reportable Occurrence No. LER 96-003

Dear Sirs:

As defined by 10 CFR 50.73, we are reporting the attached Reportable Occurrence as LER 96-003.

Very truly yours,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Robert J Wanczyk

Robert J. Wanczyk
Plant Manager

cc: Regional Administrator
USNRC
Region I
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NRC Form 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION LICENSEE EVENT REPORT (LER)	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98 ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20566-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1) VERMONT YANKEE NUCLEAR POWER STATION	DOCKET NUMBER ( ) 05000271	PAGE (3) 01 OF 03

TITLE (4) Removed Reactor Shield Blocks during Power Operations to Facilitate Outage Scheduling due to a Lack of Procedural Guidance

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 NO.(S) 05000
01	31	96	96	-- 003 --	00	03	01	96	N/A	

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: CHECK ONE OR MORE (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	100		20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		(Specify in Abstract below or in NRC Form 366A)
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)	
NAME ROBERT J. WANCZYK, PLANT MANAGER	TELEPHONE NO. (Include Area Code) 802-257-7711

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
NA					NA				
NA					NA				

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MO	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 1/31/96, as a result of a self-evaluation, it was determined that in 1990 and 1992 all three sets of reactor overhead shield blocks were removed while the reactor was at approximately 10% and 25% power respectively to facilitate outage scheduling. With the shield blocks removed at power, the Technical Support Center, used as a center to facilitate management of an emergency event, could be subjected to a dose in excess of that allowed by NUREG 0737.II.B.2.

In accordance with NUREG 0737.II.B.2, Technical Support Center (TSC) personnel cannot receive greater than a 5 REM dose for the 30 day period following a design basis accident. With all the reactor shield blocks removed, it is conservatively estimated that the highest dose could potentially be greater than 5 REM. This would be outside the design basis of the plant and contrary to the requirements of NUREG 0737.II.B.2.

The root cause of this event is under investigation. The apparent cause is attributed to a lack of formal procedural guidance governing the removal of the shield blocks. If the root cause is different from the apparent cause we will present it in a supplemental LER.

Corrective action was previously taken to revise the procedure that controls refueling preparations to include guidance that will only allow one layer of reactor shield blocks to be removed prior to placing the Reactor Mode Switch in Shutdown or Refuel position.

These are historical events which did not result in any condition that affected the health and safety of the public.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REV #	
VERMONT YANKEE NUCLEAR POWER CORPORATION	05000271	96	-- 003 --	0	02 OF 03

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

DESCRIPTION OF EVENT

On 1/31/96, as a result of a self-evaluation, it was determined that in 1990 and 1992 all three sets of reactor shields blocks (EIIS=BLK) were removed prior to the ractor being fully shutdown. The reactor shield blocks are removable concrete blocks installed above the Drywell head that serve primarily as a biological shield. There are three sets of blocks, each divided in half. Each set of blocks is approximately two feet thick which provides a total of approximately six feet of shielding. These blocks must be removed each refueling outage to allow access to the Drywell head.

The third set of reactor shield blocks were removed while the reactor was being shutdown just prior to the 1990 and 1992 outages. During these times, reactor power was approximately 10% and 25% respectively. During removal of the shield blocks, dose rate measurements were taken to determine the radiation levels on the refuel floor with the blocks removed. At that time, no studies were completed that addressed the effect on the TSC relative to NUREG 0737.II.B.2 requirements.

CAUSE OF EVENT

The apparent cause of this event is a lack of procedural guidance governing removal of the shield blocks and the impact of shielding modifications on radiological vital areas. The root cause of this event is presently under investigation. If the root cause is different than the apparent cause, it will be presented in a supplemental LER.

ANALYSIS OF EVENT

NUREG 0737.II.B.2 requires that the dose for personnel in the TSC following a design basis accident does not exceed 5 REM for the 30 day period following the postulated accident. Contrary to this, with all reactor shield blocks removed at power, the conservatively estimated dose to personnel in the TSC could potentially exceed the maximum of 5 REM for the thirty day period following the accident.

The dose to personnel would be controlled within the limits specified by NUREG 0737.II.B.2. by station procedures which require habitability assessments. Actions taken would include, adjusting the staffing schedules to ensure that the higher dose rates are considered during assignment of personnel, utilization of additional qualified personnel from other resources, and relocation of the facility if required.

Although the potential for an increase in dose existed for the short period of time that the reactor shield blocks were removed while at power, no accident occurred and therefore there was no danger to the health and safety of the public.

It should be noted that the shield blocks were removed at low power levels while the reactor was being shutdown for refueling. The resulting dose rates from a design basis accident at low power levels would be less than if an accident were to occur at higher power levels.

A study completed in 1995 assessed the dose to the TSC with only two sets of the reactor shield blocks removed. This was a conservative assessment that determined that the dose to the TSC would be approximately 4.96 REM for the thirty day period following a design basis accident. While a detailed assessment has not been completed to determine the dose with the three sets of shield blocks removed, it is conservatively estimated that the does would be greater than the 5 REM criteria in NUREG 0737.

CORRECTIVE ACTIONS

Review of potential corrective actions identified that corrective action were presiously addressed by revision of the refueling preparation procedure to allow the removal of only one set of shield blocks until the Reactor Mode Switch is in the Shutdown or Refuel position.

Should the root cause investigation in progress identify a different cause than currently identified which require additional actions a supplement to this LER will be provided.

ADDITIONAL INFORMATION

No similar conditions have been reported to the Commission within the past five years.