VERMONT YANKEE
NUCLEAR POWER CORPORATION

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June 2, 1997
BVY-97-076

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

Reference: (a) License No. DPR-28 (Docket No. 50-271)

Subject: Reportable Occurrence No. LER 97-012, Rev. 0.

As defined by 10CFR50.73, we are reporting the attached Reportable Occurrence as LER 97-01, Rev. 0.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Gregory A. Maret
Plant Manager

IE22

c: USNRC Region 1 Administrator
USNRC Resident Inspector - VYNPS
USNRC Project Manager - VYNPS

NRC Form 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
LICENSEE EVENT REPORT (LER)	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)	DOCKET NUMBER ( )	PAGE (3)
VERMONT YANKEE NUCLEAR POWER STATION	05000271	01 OF 03

TITLE (4) Residual Heat Removal Service Water Flow Could Be Potentially Less than the Design Basis Flow due to Instrument Inaccuracies

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
05	02	97	97	-- 012 --	00	06	02	97	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)	0		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	TELEPHONE NO. (Include Area Code)
GREGORY A. MARET, PLANT MANAGER	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
X	YES (If yes, complete EXPECTED SUBMISSION DATE)		NO		08	29	97

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

On 5/2/97 at 1300 hours, during the AE inspection preparation it was determined that the Residual Heat Removal(RHR) Service Water(SW) flow rate during a LOCA could potentially have been below the required design flow. The design flow of the RHRSW heat exchanger and required flow for a LOCA is 2700 Gallons Per Minute (GPM). Instrument accuracy for flow indication is +/- 200 GPM, which could have resulted in an actual RHRSW flow rate as low as 2500 GPM. This condition has been evaluated and it was determined that the RHRSW system can meet its design cooling capacity provided that the plant is only operated if river water temperature (primary cooling medium) is equal to or less than 70 degrees F. Additionally, if flow for the RHRSW system was diverted from other loads such that the actual heat exchanger flow was 2900 GPM, with instrument inaccuracy in the conservative direction, it was determined that the remaining loads would still have the required amount of cooling with the river water temperature restriction in use.

The root cause investigation is in progress. A supplemental LER will be submitted once the root cause has been determined. Immediate corrective actions included the initiation of an Event Report to document the concern and notify the Nuclear Regulatory Commission (NRC), the initiation of a Basis for Maintaining Operation (BMO) document with a mandatory read and sign form for the Operations on-shift crews, and the establishment of a river water temperature administrative limit of 70 degrees F. Since plant start-up there has been no accident or LOCA conditions that would have required the use of the RHRSW system in the accident mode. Analysis of the event shows that the potential low flow can be augmented using the opposite RHR loop. There was no threat to the health and safety of the public and no safety consequences resulted from this event.

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

DESCRIPTION OF EVENT

On 5/2/97 at 1300 hours, during an AE inspection preparation, it was determined that the Residual Heat Removal(RHR)(EIIS=BO) Service Water(SW)(EIIS=BI) flow rate during a LOCA could potentially have been below the required design flow. The design flow of the RHRSW heat exchanger is 2700 Gallons Per minute (GPM), the same flow rate which is required by the LOCA analysis. To prevent exceeding the RHRSW heat exchanger design flow rate, plant procedures direct the operators to limit the RHRSW heat exchanger flow rate to the design limit of 2700 GPM. This limit would be imposed using available indication; however, instrument accuracy for flow indication is +/- 200 GPM which could have resulted in an actual RHRSW flow rate as low as 2500 GPM.

This condition has been evaluated and it was determined by engineering analysis that the RHRSW system can meet its design cooling capacity, with the reduced flow of 2500 GPM, provided that plant operation only continue if river water temperature (primary cooling medium) is equal to or less than 70 degrees F. Additionally, if flow for the RHRSW system was diverted from other loads such that the actual flow was 2900 GPM, to account for instrument inaccuracy in the conservative direction, it was determined that the other loads would still have the required amount of cooling provided the river water temperature restriction was maintained. There would be no adverse impact on the heat exchanger as a result of the increased flow.

CAUSE OF EVENT

The root cause of this event is under investigation. A supplemental License Event Report will be submitted once the root cause has been determined.

ANALYSIS OF EVENT

The RHRSW System provides a dynamic heat sink for the RHR System by supplying sufficient cooling capacity during a design basis accident (DBA) and minimizes the probability of a release of radioactive contaminants to the environs.

The design basis LOCA parameters for the RHRSW system are: a flow of 2700 GPM, initial torus temperature of 90 degrees F, and a maximum river temperature of 85 degrees F. This allows the torus water to be maintained at or below 176 degrees F following a LOCA. The maximum Torus water temperature of 176 degrees F ensures that there is sufficient Net Positive Suction Head (NPSH) for the Emergency Core Cooling System (ECCS) pumps and prevents degradation of Environmentally Qualified equipment. During a LOCA, the Core Spray system would be used for injecting water into the reactor vessel and the one or both RHR loops could be used for Torus cooling.

Instrument accuracies were not initially considered when determining design basis parameters. The flow indication accuracy is +/- 200 GPM. Subsequently, if the RHRSW flow, to cool the Torus during an accident, was set at 2700 GPM the potential exists for the actual flow to be 2500 GPM. This is less flow than assumed in the analysis of record which demonstrates Vermont Yankee's ability to maintain the Torus water at or below the required temperature with river water temperature at or near 85 degrees F. If this had happened, operators would have been alerted and given appropriate direction by plant Emergency Operating Procedures (EOP's) using the parameters available in the Control Room which would have indicated the adverse trend in Torus water temperature. Options available to the operators would be to place both RHR loops on Torus cooling, using the Core Spray System to fill and maintain the reactor vessel cooling. Engineering evaluations would have been provided by the Technical Support Center.

Therefore, adequate cooling was available for the Torus and there was no threat to the health and safety of the public.

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

CORRECTIVE ACTIONS

Immediate Corrective Actions.

1. An Event Report to document the event was written.

2. The Nuclear Regulatory Commission (NRC) was notified in accordance with 10 CFR 50.72.

3. A Basis for Maintaining Operation (BMO) document was written.

4. A mandatory read and sign form requiring the Operations on-shift crews to read and understand the BMO was initiated.

5. An administrative river water temperature limit of 70 degrees F was established.

Long Term Corrective Actions

1. The long term corrective action will evaluate the current Service Water flow model at the upper limits for river temperature and flow under the conditions assumed for the RHR System operation. This will be completed by 7/1/97.

ADDITIONAL INFORMATION

During the past five years similar events involving original design specifications have been reported as LER's 93-13, 95-02, 97-02, and 97-06.