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March 18, 1994

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 93-018, Supplement 1

Dear Sirs:

As defined by 10 CFR 50.73, we are reporting the attached Reportable Occurrence as LER 93-018, Supplement 1.

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 U.S. NUCLEAR REGULATORY COMMISSION (6-89)	APPROVED OMS NO. 3150-0104 EXPIRES 4/30/92 ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-350), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3160-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20603.

FACILITY NAME (1)	DOCKET NO. (2)	PAGE (3)
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TITLE (4)

Group 4 Primary Containment Isolation on Initiation of "A" Shutdown Cooling System due to Pressure Spike

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQ #	REV #	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NO. (S)
1 2	1 7	9 3	9 3	- 0 1 8 -	0 1	0 3	1 8	9 4		0 5 0 0 0
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OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO REQ'MTS OF 10 CFR §: CHECK ONE OR MORE (11)							
			20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	0 0 0		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:
....			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 NO.	
	AREA CODE	
ROBERT J. WANCZYK, PLANT MANAGER	8 0 2	2 5 7 - 7 7 1 1

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

CAUSE	SYST	COMPONENT	MFR	REPORTABLE TO NPRDS		CAUSE	SYST	COMPONENT	MFR	REPORTABLE TO NPRDS	
N/A					....	N/A					....
N/A					....	N/A					....

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MO	DAY	YR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approx. fifteen single-space typewritten lines) (16)

On 12/17/93 at 1844 hours, with the Reactor Shutdown and vessel cooldown in progress for repairs to the main condenser to correct inleakage, and with the "A" loop Residual Heat Removal (RHR) system flushed and lined up for Shutdown Cooling (SDC), a Group 4 Primary Containment Isolation Signal (PCIS) was received while attempting to start the "A" RHR pump. The Group 4 Isolation signal resulted in a trip of the "A" RHR Pump and closure of Shutdown Cooling Isolation Valves. The Group 4 Isolation signal was reset and the "A" RHR pump was successfully restarted for shutdown cooling at 1857. The cause of this event is a pressure spike caused by the collapse of voids formed between the Injection valve and the downstream check valve upon startup of the "A" RHR pump. The Root Cause has been determined to be in the original design specification in that no high point vent was factored into the RHR SDC loop to address voids in the piping. An engineering evaluation and a review of piping an vessel thermal analyses has resulted in a proposed procedural changes for an alternate fill path from the condensate transfer system which will eliminate voids prior to system startup. Supplemental instrumentation will be installed on the RHR SDC system piping to monitor the effectiveness of the proposed procedure change.

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

DESCRIPTION OF EVENT

On 12/17/93 at 1844 hours, with the Reactor Vessel cooldown in progress following a Reactor shutdown for repairs to correct condenser air inleakage, and with the "A" loop Residual Heat Removal (RHR) (BO*) system flushed with 5500 gallons of water and lined up for Shutdown Cooling, a Group 4 Primary Containment Isolation Signal (PCIS) (JM*) was received while attempting to start the "A" RHR pump. The Group 4 Isolation signal resulted in a trip of the "A" RHR Pump and closure of the Shutdown Cooling Isolation Valves. The Group 4 Isolation signal was reset, an additional 1000 gallon hot flush was performed, and the "A" RHR pump was successfully restarted for shutdown cooling at 1857 hours.

(* = EIIS COMPONENT IDENTIFIER)

Interviews with Operations personnel indicated that after the first hot flush with the outboard injection valve throttled open slightly and pressure stabilized at 70 to 80 psig in the loop, an audible sound was noted in the pipe and a Group 4 isolation occurred approximately 4 seconds after the pump start. The initiators for a Group 4 Isolation are Reactor low water level, high Drywell pressure, or high RHR system pressure. A review of plant parameters indicated that Reactor water level and Drywell pressure were acceptable and that the RHR system pressure was well below the initiation set point of 130 psig for PS-2-128A/B. At the time of the isolation Reactor pressure was approximately 55 psig. As the injection valve was throttled open, RHR system pressure equalized and a Reactor water level drop of approximately 3 inches was noted.

A similar event was identified in LER 91-006 at which time the cause was identified as a section of the RHR piping between the inboard check valve and the outboard injection valve remaining depressurized following the completion of the "B" loop "hot" flush prior to the system initiation. In both events this depressurized section of piping is believed to be responsible for the pressure surge in the RHR piping which resulted in the system isolation.

Procedural changes had been made following the event in March of 1991. The procedural changes resulted in throttling open the outboard injection, RHR-27 A(B) valve prior to starting the RHR pump. The purpose of doing this was to allow the piping downstream of the valve to refill/repressurize prior to starting the pump and circulating water to the reactor. During the 1992 Refueling Outage, the procedural changes appeared to be successful. However, the test instrumentation still showed small spikes in system pressures when the system was started. Subsequent to the 1992 outage procedural changes, a concern was raised as to the amount the RHR 27 A(B) valve could be opened before subjecting the system to excessive flow when the RHR pump was started. A subsequent procedural change was made to address this concern and reduced the time the valve was throttled open prior to pump start from 3 seconds to 1 second. This change in throttling time now appears to be insufficient in both RHR loops to allow the piping to fully re-pressurize and results in a pressure surge of sufficient magnitude, when establishing Shutdown Cooling pump flow, to cause the Group 4 isolation.

At the time of this event all earlier events had occurred on the "B" RHR SDC loop. An engineering evaluation was in progress as a corrective action from the previous events to determine the cause of the isolation and recommend additional corrective actions. As a result of this occurrence, the evaluation was broadened to include both "A" and "B" loops and was expedited by plant management.

CAUSE OF EVENT

The Root Cause of this event is attributed to the lack of a high point vent on this section of RHR piping. Pressure spikes resulting from void collapse upon system start up were not anticipated in the original system piping design. Since there is no high point vent and based on vessel water level drop data it appears that opening the V10-27A/B valve is not completely filling the piping section. Pressure is equalizing and compressing the air/vapor volume but not completely eliminating the void in the piping. The subsequent pump start and rapid further opening of the V10-27A/B valve results in a pressure spike due to void collapse. This resultant short duration pressure transient which is characteristic of a void collapse is of sufficient magnitude to initiate a PCIS Group 4 isolation signal.

ANALYSIS OF EVENT

The events detailed in this report had minimal safety implications.

1. The Primary Containment Isolation System (PCIS) operated as designed and isolated Shutdown Cooling upon sensing a high pressure.

2. Reactor depressurization and cooldown was maintained during this event using the Main Condenser. All Emergency Core Cooling Systems were available, if required.

3. Shutdown Cooling is used only when the Reactor is at 0% power. Therefore this same Group 4 Isolation could not have impacted plant safety while the plant was operating.

4. The "B" train of Shutdown Cooling was not in service but was available to be used if needed.

An engineering evaluation has been completed utilizing test data from previous "B" loop temporary instrumentation, Plant Operating Procedures, GE Service Information Letters, previous License Event Reports, Electric Power Research Institute report data, several other Nuclear plants experience with SDC and our RHR system piping configuration. The evaluation concluded that the existing piping configurations and a lack of adequate high point venting provisions and associated procedures are contributors to the RHR system pressure surges which initiate the Group 4 Isolation. With the inboard check valves V10-46A/B at a high point or hump in the piping, and no venting provisions available, the scenario for a trapped steam void collapse upon initiation of SDC is created.

The most likely location for a void is between the V10-27A/B and V10-46A/B valves. There is approximately 100 cubic feet of pipe between these two valves and an elevation difference of 17 feet. This section of piping is a high point in the system and does not have a high point vent and is depressurized during periodic testing while the plant is in operation. Test data indicates that prior to opening the injection valve following the hot flush, a low pressure exists in this section of piping. After opening the injection valve a pressure increase is noted. Although the pressure approaches the overall system pressure, the volume represented by the typical reactor water level drop observed when the valve is opened is less than the pipe volume. This implies the pipe is not completely filling with liquid. A second and typically larger water level drop occurs when the pump starts, indicating the volume is filling. The very small or undetectable water level drop during a second pump start would imply the piping had been substantially filled.

The engineering evaluation investigated overpressures that may result from a void collapse in the RHR return line where the PCIS Group 4 isolation instrumentation is located. Various initial RHR flow rates were used to predict pressures experienced at the PCIS initiation instrumentation resulting from a void collapse using the more severe assumption of a void upstream of V10-46A/B. The calculation demonstrates that overpressure intensity, in this case, is directly related to RHR flow and that for lower initial flow rates, less than approximately 2000 gpm, the resulting pressure transient is expected to be below the isolation trip setpoint. Flow data from this event indicates that a 4000 gpm flow rate was present at the time of the isolation which is capable of producing an isolation signal if voids are present in the piping.

An alternate approach for assuring the pipe is completely filled prior to RHR system startup would be to use the condensate transfer system. This approach would open V10-27A/B and flush to the vessel by opening V10-70A/B and V10-71A/B until a vessel level increase was observed. After securing V10-70A/B and V10-71A/B, SDC would then be initiated per the current procedure. This should be a more controllable process than the existing practice of slightly opening V10-27A/B, as it is difficult to determine the valve position (and flow) due to a lack of a precise position feedback mechanism.

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

The use of the condensate transfer system for filling of the RHR piping introduces water into the system at a lower temperature than that currently used for Shutdown Cooling flushing. Therefore, the piping and vessel analyses of record have been evaluated and it has been confirmed that the use of this procedure will not exceed the number of equivalent full temperature cycles for which these components are currently analyzed.

CORRECTIVE ACTIONS

Immediate corrective actions:

1. Reset the PCIS isolation, perform an additional 1000 gal hot flush, and restoration of Shutdown Cooling on the "A" loop of RHR.

2. Observations by the Auxiliary Operator stationed at the outboard injection valve area noted that there was no visible pipe movement during the event. An Operations walkdown of the system piping did not indicate any problems.

Long term corrective actions:

1. A Significant Corrective Action report (CAR 94-01) has been completed and has resulted in an engineering evaluation which has been approved by plant management. The accepted recommendation includes modifying procedures to fill the process piping from the condensate transfer system by opening V10-70A/B and V10-71A/B until either a vessel level increase is noted, or other indication is observed that verifies the RHR SDC loop is completely filled. After securing V10-70A/B and V10-71A/B, SDC would then be initiated per the current Operations procedure. Following completion of a formal Safety Evaluation on the new flushing technique and baring any unforseen issues resulting from that evaluation it is our expectation to implement the new technique by July 1994.

2. In order to monitor the effectiveness of these procedural changes, supplemental instrumentation used in previous data acquisitions will be re-installed on the RHR SDC system in support of the new technique to monitor the system parameters and verify adequacy of the corrective actions implemented.

ADDITIONAL INFORMATION

There have been similar events reported to the Commission by Vermont Yankee as LER 91-006 and LER 93-011.