CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT
362 INJUN HOLLOW ROAD • EAST HAMPTON, CT 06424-3099

April 5, 1994
Re: 10CFR50.73(a)(2)(v)

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

Reference: Facility Operating License No. DPR-61
Docket No. 50-213
Reportable Occurrence LER 50-213/94-007-00

Gentlemen:

This letter forwards the Licensee Event Report 94-007-00, required to be submitted, pursuant to the requirements of the Haddam Neck Plant's Technical Specifications.

Very truly yours,

John P. Stetz
Vice President

JPS/mlg

Attachment: LER 50-213/94-007-00

cc: Mr. Thomas T. Martin
Regional Administrator, Region I
475 Allendale Road
King of Prussia, PA 19406

William Raymond
Sr. Resident Inspector
Haddam Neck

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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): Haddam Neck

DOCKET NUMBER (2): 0 5 0 0 0 2 1 3

PAGE (3): 1 OF 0 7

TITLE (4): Potential for Radiological Release During Post LOCA Sump Recirculation

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 3	0 8	9 4	9 4	0 0 7	0 0	0 4	0 5	9 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[X] 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 (AREA CODE)
J. Majewski, Senior Engineer	2 0 3 2 6 7 - 2 5 5 6

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

ABSTRACT

On March 8, 1994, at approximately 1320 hours, with the plant in Mode 5 (cold shutdown), it was determined that the potential existed to lift a relief valve in the Chemical and Volume Control System (CVCS) during the performance of the emergency response procedure to establish containment sump recirculation following a Loss of Coolant Accident (LOCA). This in turn could cause another relief valve to lift in the Radioactive Waste Gas System which discharges to the environment via the plant stack. If such an event were to occur, radiological dose consequences could be more severe than previously predicted. The cause of this condition was a failure to recognize the full potential adverse impacts of a change to an emergency response procedure during the review of the procedure change in 1990. Corrective action involved revising the affected emergency response procedure to eliminate the potential overpressure condition. Also, in October, 1991 the station procedure addressing the preparation, change, and review of emergency response procedures was revised to provide for a more rigorous technical review of new or changed procedures than had existed previously.

NRC Form 366 (9-83)

BACKGROUND INFORMATION

Several systems can be used at the Haddam Neck plant for containment sump recirculation following a Loss of Coolant Accident (LOCA) (see Figure 1). Namely, the Residual Heat Removal (RHR) system (EIIS code BP), the High Pressure Safety Injection (HPSI) system (EIIS Code BQ), and the charging portion of the Chemical and Volume Control system (CVCS) (EIIS Code CB). After the Refueling Water Storage Tank has been emptied to a specified low level during the injection phase, the RHR pumps and heat exchangers are then used to remove water accumulated in the containment sump and discharge it back to the RCS via the Charging and/or HPSI system.

The Charging system consists of three pumps in parallel (two moderate capacity centrifugal and one low capacity positive displacement) which discharge into a common header. Suction can be supplied from the Volume Control Tank (VCT), the Refueling Water Storage Tank (RWST), or the discharge of the RHR pumps and heat exchangers via either of two parallel motor operated valves (RH-MOV-33A and 33B). The discharge header splits into two paths, one provides Reactor Coolant Pump seal water supply and the other provides RCS makeup supply via the charging header to the RCS loop two cold leg. The discharge of the positive displacement pump is equipped with a relief valve (CH-RV-280) which is designed to protect the downstream piping and components from overpressure if the discharge flow path were to be isolated with the positive displacement pump in operation.

CH-RV-280 relieves to the Primary Drains Tank (PDT) in the radioactive waste gas system (EIIS Code:WE). The PDT is a 7500 gallon tank that receives waste water containing dissolved hydrogen or fission gases from various reactor plant sources . Liquid in the PDT is pumped by one of two parallel pumps through a filter and a heater to the degasifier where it is sprayed through nozzles causing the heated liquid to flash thereby releasing the dissolved gasses. The degasified liquid is pumped through a cooler either to the CVCS or to outdoor storage tanks for eventual release offsite. The gases stripped from the liquid are cooled to condense any steam and are then routed to the waste gas surge tank. The waste gas surge tank receives radioactive waste gases from various sources including the gas space of the PDT. Collected gases are compressed by one of two parallel compressors and stored in three waste gas decay tanks for eventual release off site. The waste gas surge tank is protected against overpressure by an air operated valve actuated by a pressure switch at 14 psig and also by a spring loaded relief valve set at 20 psig. Both of these valves relieve directly to the environment via the plant stack.

Overpressure protection for each of the three waste gas decay tanks is provided by an air operated valve actuated by a pressure switch at 215 psig and a spring loaded relief valve set at 225 psig. Both of these valves relieve directly to the environment via the plant stack.

EVENT DESCRIPTION

On March 8, 1994, at approximately 1320 hours, with the plant in Mode 5 (cold shutdown), an engineering evaluation determined that the potential existed to lift a relief valve in the Chemical and Volume Control System (CVCS) during the performance of the emergency response procedure to establish containment sump recirculation following a Loss of Coolant Accident (LOCA). Emergency response procedure ES-1.3, "Transfer to Sump Recirculation" provides for the use of the RHR, Charging, and HPSI pumps for sump recirculation if offsite power is available to the emergency buses. The RHR pumps are aligned to take suction from the containment sump and discharge to the suction of the charging pumps. One charging pump is checked to be running or is started and the flow control valves and motor operated isolation valves on the charging header are checked to be open. Flow from the sump to the RCS is then initiated by opening the RHR to Charging pump suction motor operated isolation valves (RH-MOV-33A and 33B). The charging pump suction is then isolated from the VCT and the RWST. If RCS pressure is less than 1500 psig, actions are then taken to align the discharge of the RHR pumps to the suction of the HPSI pumps. After this alignment is completed, the motor operated isolation valves on the charging header (CH-MOV-292B&C) are closed and one HPSI pump is started. The charging header isolation valves are closed to prevent excessive flow and runout of the RHR pumps once the HPSI pumps are started. The charging pump is left running to inject water through the RCP seals.

The reduced charging pump flow resulting from the closure of the charging header isolation valves when combined with the discharge pressure from the RHR pumps results in a charging pump discharge pressure in excess of the lift setpoint of CH-RV-280. CH-RV-280 has a lift setpoint of 2735 psig ± 3% (2653 to 2817 psig). The charging pump discharge pressure during such an event was estimated to be approximately 2658 psig, which slightly exceeds the lowest allowable lift setpoint value for CH-RV-280. CH-RV-280 has a rated capacity of 30 gpm (the maximum flow rate of the positive displacement charging pump) at an accumulated pressure of 3300 psig. Thus, at a pressure very close to the lift setpoint, the flow rate would be expected to be much less than 30 gpm.

SAFETY ASSESSMENT

The potential to lift CH-RV-280 during post LOCA sump recirculation is reportable under 10 CFR 50.73 (a)(2)(v)(C) as a condition that alone could have prevented the fulfillment of the safety function of a system that is needed to control the release of radioactive material.

This event is only potentially significant following a large break LOCA in which significant core damage has occurred. If CH-RV-280 were to lift during sump recirculation using the charging pumps, there would be a slight reduction in the total coolant injection flow. As much as 30 gpm could be potentially lost, although much less than this would actually be expected as discussed above. Thus, the effect of CH-RV-280 lifting on the ability to maintain core cooling is insignificant.

A second concern with the potential to lift CH-RV-280 during post LOCA sump recirculation would be the increase in system leakage outside containment, that could result in an increase in calculated offsite dose consequences. The radiological consequence analysis for large break LOCA assumes a total system leakage of 3 liters per hour, which is significantly less than what could potentially be released. However, CH-RV-280 is downstream of the RHR heat exchangers and the temperature of the recirculated coolant leaked at this point would be less than 200 degrees (F). At this temperature, no flashing of the coolant will occur and the primary mechanism for iodine to become airborne will not be present. Thus, the actual airborne iodine fraction would be significantly less than assumed in the Final Safety Analysis Report (FSAR) radiological analysis. There is the possibility that the coolant released to the plant stack could leak out onto the ground and eventually get offsite. It would take a substantial amount of time for a leak of this magnitude to reach the site boundary and could reasonably be expected to be discovered by radiation monitoring surveys conducted during an accident. Based on the above, the actual safety significance is judged to be moderate.

CORRECTIVE ACTION

Corrective action involved revising emergency response procedure ES-1.3 to eliminate the potential overpressure condition by stopping the running charging pump prior to starting the HPSI pump. A review of the sump recirculation flow path was performed to determine if there were any other system relief valves that could potentially lift during performance of ES-1.3 and none were found.

Relief flow from CH-RV-280 would cause the level in the PDT to increase, eventually starting an automatic degasifier cycle to process the water in the PDT. The PDT, degasifier, and waste gas systems would cycle on and off automatically to process the input to the PDT provided that steam is available to heat the degasifier feedwater. Unavailability of process steam would result in the failure of the PDT pumps to start since a temperature switch on the degasifier necessary for PDT pump start would not be actuated. PDT level would continue to increase. Eventually, the PDT and waste gas surge tank would become filled to capacity. Pressure up to this point has been controlled by the automatic starting and stopping of the waste gas compressors. Eventually, the suction lines for the waste gas compressors would be filled with water. If the temperature of the water is 125 degrees (F) or greater, the compressors will trip on high suction temperature. It is also possible that the compressors may trip on overload. Either of these two conditions will result in waste gas surge tank pressure reaching the setpoint of the relief valves and a radiological release to the environment via the plant stack. If the compressors continue to run, the waste gas decay tanks would eventually fill and be pressurized. Automatic alignment to the next unpressurized tank would occur until all three tanks had been pressurized. Pressure in the last tank would continue to increase to the lift setpoint of the relief valves also resulting in a radiological release to the environment via the plant stack.

Indications and alarms for the PDT, waste gas surge tank, degasifier, and waste gas compressors are provided locally on a control panel in the primary auxiliary building. However, radiological conditions in this building during sump recirculation may not permit monitoring of this control panel. Radiation monitoring for the plant stack which is displayed in the control room would provide the control room operators with information on a release in progress. This information alone, however, would not be sufficient to allow diagnosis and mitigation of such a release.

CAUSE OF THE EVENT

The cause of this condition was a failure to recognize the full potential adverse impacts of a change to emergency response procedure ES-1.3 during the review of the change in 1990. The alignment for sump recirculation using a combination of HPSI and Charging pumps was added to procedure ES-1.3 in April, 1990 following modifications to the HPSI pump suction lines which added valves to accomplish the alignment of the RHR pump discharge to the suction of the HPSI pumps.

Also, in October, 1991 the station procedure (ACP 1.2-6.13, "Emergency Response Procedure Generation Guidelines" which addresses the preparation, change, and review of emergency response procedures was revised to provide for a more rigorous technical review of new or changed procedures than had existed previously.

ADDITIONAL INFORMATION

N/A

PREVIOUS SIMILAR EVENTS

LER 90-011-00

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Haddam Neck	0 5 0 0 0 2 1 3	9 4	— 0 0 7 —	0 0	0 7 OF 0 7

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

Figure 1