OPPD
Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247

September 30, 1998
LIC-98-0126

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Reference: Docket No. 50-285

Subject: Licensee Event Report 1998-009 Revision 1 for the Fort Calhoun Station

Please find attached Licensee Event Report 1998-009 Revision 1 dated September 30, 1998. This report is being submitted pursuant to 10CFR50.73(a)(2)(ii)(B). If you should have any questions, please contact me.

Sincerely,

Gambhir

S. K. Gambhir
Division Manager
Nuclear Operations

EPM/epm

Attachment

c: E. W. Merschoff, NRC Regional Administrator, Region IV
L. R. Wharton, NRC Project Manager
W. C. Walker, NRC Senior Resident Inspector
INPO Records Center
Winston and Strawn

IE22

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

APPROVE[illegible] OMB NO. 3150-0104
[illegible]ES 4/30/98

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Fort Calhoun Station Unit No. 1	05000285	1 OF 7

TITLE (4)
Waste Disposal System Containment Isolation Valves Outside Design Basis

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 05000
07	22	1998	1998	-- 009 --	01	09	30	1998	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 1
POWER LEVEL (10): 100

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)
	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 NUMBER (Include Area Code)
Kevin C. Hyde, Senior Nuclear Design Engineer, Mechanical	402-533-6571

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
B	IP	ISV	H195	Y					

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	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 June 25, 1998, the Post-Accident Sampling System (PASS) return valve to the Reactor Coolant Drain Tank (RCDT) failed to open while running the PASS. This resulted in the upstream Waste Disposal System (WDS) Containment Isolation Valve, HCV-500A, experiencing a body-to-bonnet leak. The PASS sequence being run involved drawing a liquid Reactor Coolant System sample in the accident mode and discharging to the WDS. HCV-500A was declared inoperable and shut per the plant Technical Specifications. HCV-500A was subsequently repaired, tested and returned to service on June 26, 1998. The PASS return valve to the RCDT has also been repaired.

While performing a failure analysis of HCV-500A a design deficiency with the PASS interface with the WDS was discovered. The PASS is capable of discharging a sample at 450 degrees Fahrenheit (F) and 2100 psig. The WDS Containment Isolation valves are rated at 300 degrees F (maximum) at 90 psig and 120 degrees F at 150 psig. Therefore, the PASS has the potential to exceed the design basis of the WDS Containment Isolation valves. This condition is outside the design basis of the plant.

A modification is being completed to correct the problem.

FACILITY NAME (1)	DOCKET	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Fort Calhoun Station Unit No. 1	05000285	1998	- 009 -	01	2 OF 7

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

BACKGROUND

The basic function of the Post Accident Sampling System (PASS) (EIIS: IP) is to provide the capability to obtain and analyze samples of reactor coolant and containment atmosphere under design basis accident conditions with a degraded core while minimizing personnel radiation exposure. The sample analysis provides indication of the degree of core damage and provides information to monitor reactor conditions after an accident.

The PASS also provides the capability to obtain and analyze samples of reactor coolant and containment atmosphere during normal operation and to monitor reactor conditions.

Liquid samples are introduced into the system from the Reactor Coolant System (RCS) (EIIS: AB) or the Low Pressure Safety Injection (LPSI) (EIIS: BP) pump discharge. The pressure of the RCS is used whenever possible as the driving force to overcome system pressure drop. If the primary system is depressurized, then LPSI samples must be used.

The waste disposal pipe is 151R classification which has a design rating of 240 pounds per square inch gage (psig) at 200 deg Fahrenheit (F), 210 psig at 300 degrees F and 150 psig at 500 degrees F. HCV-500A and B (EIIS: ISV) are designed for a rating of 150 psig at 120 degrees F and 90 psig at 300 degrees F. The temperature rating of the valve diaphragm is 300 degrees F.

EVENT DESCRIPTION

On June 25, 1998, the PASS return valve to the Reactor Coolant Drain Tank (RCDT) (EIIS: TK), HCV-6743 (EIIS: ISV) (see Figure 1), failed to open while running the PASS. HCV-6743 failing closed pressurized the line from the PASS system to HCV-6743. This resulted in the upstream Waste Disposal System (WDS) (EIIS: WH) Containment Isolation Valve, HCV-500A, experiencing a body-to-bonnet leak. The PASS sequence being run involved drawing a liquid RCS sample in the accident mode. The PASS discharges to the WDS. HCV-500A was declared inoperable and shut per the plant Technical Specifications (TS) as required. HCV-500A was subsequently repaired, tested and returned to service on June 26, 1998. HCV-6743 has also been repaired.

In the course of performing a failure analysis in conjunction with resolving the issue, a design deficiency with the PASS interface with the WDS was discovered. The PASS is capable of discharging a sample at 450 degrees F and 2100 psig. The WDS Containment Isolation valves are rated at 300 degrees F (maximum) at 90 psig and 120 degrees F at 150 psig. Therefore, the PASS has the potential to challenge the integrity of the WDS Containment Isolation valves.

The discharge pressure of the PASS drain pump SL-41 (EIIS: P) was locally observed to be 250 psig. The PASS RCS sample is run through sample heat exchanger (SL-3), which is cooled by Component Cooling Water (CCW). The discharge temperature was not observed while running the PASS sample. The design rating of SL-3 is to cool a sample from 643 degrees F to 120 degrees F

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Fort Calhoun Station Unit No. 1	05000285	1998	- 009 -	01	3 OF 7

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

at a design flow rate of 0.6 gallons per minute (gpm). The 0.6 gpm flow rate is for the design basis LOCA when the system is at a relatively low pressure. The PASS sample is run at a higher flow rate than 0.6 gpm because it was run at normal RCS pressure, therefore the temperature is expected to have been higher than 120 degrees F.

On July 22, 1998, at 1708 Central Daylight Time (CDT) it was determined that this condition is outside the design basis of the plant. At 1742 CDT a non-emergency one hour report was made to the NRC Operations Center pursuant to 10CFR50.72(B)(1)(ii)(B). This report is being submitted pursuant to 10 CFR 50.73(a)(2)(ii)(B).

SAFETY SIGNIFICANCE

This event impacts the plant following either a large or small break Loss of Coolant Accident (LOCA) after which a PASS sample is collected. The following evaluation includes a review of both the Small Break Loss of Coolant Accident (SB-LOCA) and the Large Break Loss of Coolant Accident (LB-LOCA) where (1) a free flow path is available to the RCDT through HCV-6743 and (2) when the flow path is blocked.

LB-LOCA

Following a large break LOCA the RCS will be depressurized at the time a sample is taken using the PASS. The sample will be taken using Shutdown Cooling (SDC) flow from the LPSI pump discharge to drive the flow. The discharge pressure of the LPSI pumps is a maximum of 200 psi (dead head pressure) which is above the 150 psi rating of the containment isolation valves HCV-500A/B. The temperature is expected to be below the 120 degree rating of the isolation valves since the sample cooler SL-3 will be in service.

Free Flow to the RCDT: If the line to the RCDT is open, the pressure at HCV-500A/B will be equal to that of the RCDT plus any line loss associated with the piping and components leading to the RCDT. This line loss is expected to be negligible due to the low flow rate, approximately 0.6 gpm. As discussed the temperature of the fluid is 120 degrees F or less. This condition is within the rating of the containment isolation valves and the associated piping system.

RCDT flow path blocked: If the flow path to the RCDT is blocked due to a failure of HCV-500A/B or HCV-6743 to open or some other blockage, the full LPSI pressure will be applied to HCV-500A/B. This pressure is expected to produce leakage through the body-to-bonnet flange due to deformation of the installed diaphragm. The flow rate of this leakage is expected to be approximately 0.6 gpm for the duration of the sampling sequence while HCV-2574 is open. The spill does not affect accessibility to areas in the Auxiliary Building that are currently considered to be post LB-LOCA accessible.

Any airborne activity resulting from the spill will be contained and monitored by the Auxiliary Building Heating Ventilation and Air Conditions (HVAC) (EIIS:

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

AHU) system. In the event that the spill produces unacceptable releases, the Auxiliary Building ventilation will be controlled to minimize the release and isolate the affected areas per Abnormal Operating Procedure (AOP) 9 "High Radioactivity."

Containment integrity is not impacted by this event. When HCV-2574 (EIIS: ISV) closes and the pressure in the piping is reduced due to the leakage, the diaphragm in HCV-500A/B will return to its normal state and reseal the body-to-bonnet boundary. In addition, containment pressure will be significantly reduced by the time a PASS sample is taken (31 psi, 1 hr after the LOCA). HCV-500A/B are required to maintain integrity to assure containment isolation at 60 psi.

Based on the reduced containment pressure at the time the sample would be taken and the recovery of the diaphragm after depressurization, this event does not result in a significant safety concern as related to consequences associated with a LB-LOCA.

SB-LOCA

Following a SB-LOCA the RCS could be pressurized to approximately 1400 psi at the time a sample is taken using the PASS. The sample will be taken using RCS pressure to drive the flow. The pressure of the sample will be approximately 1400 psi which is above the 150 psi rating of valves HCV-500A/B. The temperature is not expected to be above the 300 degree rating of the valves. The temperature of the liquid has been conservatively calculated to be less than 300 degrees F.

Free Flow to the RCDT: If the line to the RCDT is open, the pressure at HCV-500A/B will be equal to that of the RCDT plus any line loss associated with the piping and components leading to the RCDT. This line loss is expected to be negligible due to the low flow rate. This condition is considered to be acceptable from a design pressure standpoint. Since the pressure will be low, a temperature of up to 300 degrees F is considered to be acceptable based on the rating of the diaphragm.

RCDT flow path blocked: If the flow path to the RCDT is blocked due to a failure of HCV-500A/B or HCV-6743 to open or some other blockage, full RCS pressure will be applied to HCV-500A/B. This pressure is expected to produce leakage through the body-to-bonnet flange due to deformation of the diaphragm. Design Engineering has determined that the small break LOCA does not result in failed fuel. Access to rooms 13 and 23 would therefore not be affected. That being the case, the spill does not affect accessibility to areas in the Auxiliary Building that are currently considered to be post SB-LOCA accessible.

Any airborne activity resulting from the spill will be contained and monitored by the Auxiliary Building HVAC system. In the event that the spill produces unacceptable releases, the Auxiliary Building ventilation will be controlled to minimize the release and isolate the affected areas per AOP-9.

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

Containment integrity is not considered to be impacted by this event. The pressure applied to HCV-500A/B is much higher than in the case of the LB-LOCA. However, the diaphragm is expected to leak at the body-to-bonnet flange to relieve pressure and the internal configuration is expected to prevent a rupture of the diaphragm. In addition, if the diaphragm(s) is (are) damaged, leakage from the containment atmosphere would require an additional failure of the closed loop piping associated with the RCDT. Since fuel failures are not expected for a small break LOCA, the impact of RCDT leakage to room 13 due to diaphragm failure is considered to be minimal.

If FCS had experienced a LOCA, with the PASS in its currently designed condition, and a PASS sample was taken, Auxiliary Building accessability would not have been affected, Containment integrity would not have been significantly affected and a PASS sample could have been collected. The sample would have resulted in some spillage of RCS to the floor of room 13. Due to the small volume of the spill and the operability of the Auxiliary Building HVAC and operator actions per AOP-09, offsite dose rates would not have been affected. Therefore, this event would have had little impact on the health and safety of the public.

CONCLUSION

The main concern of this event is over-pressurization of the WDS piping.

The root cause of this event is over reliance on contractor design and review of plant modifications affecting system and component design basis. In the case of this event, this resulted in an inadequate design interface with an existing system.

During the investigation associated with this LER it was determined that in 1995 HCV-500A had been over-pressurized. This event occurred during a PASS sequence which opened HCV-2504A prior to opening HCV-6743 to establish a flow path to the RCDT. This event also resulted in leakage from HCV-500A; however, the cause was determined to be a fault or error in the programmed sequence. The sequence was corrected. Comments associated with corrective action document specifically questioned the adequacy of the over-pressure protection (leak protection) of the WDS header. It was incorrectly determined at that time that over-pressure protection was provided by the relief valve installed on the RCDT.

CORRECTIVE ACTIONS

Since this modification was completed in 1981 there have been significant improvements to the design process. Current modification procedures include a multi-disciplinary modification review committee and a review of design basis using a system interaction check list. In the case of the PASS design and installation, current procedures and system interaction checklists would have prompted a review of the interface and could have prevented this condition.

The PASS reactor coolant liquid sample is administratively inoperable until the over-pressure condition on the WDS containment isolation valves is

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Fort Calhoun Station Unit No. 1	05000285	1998	- 009 -	01	6 OF 7

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

resolved. It should be noted that the severe accident management guidelines that are being developed, utilize the PASS as a method of confirming containment hydrogen concentration. This portion of the PASS is not affected by the WDS problem.

A modification to the PASS has been designed to correct this situation. The modification will be completed and operable by October 23, 1998.

PREVIOUS SIMILAR EVENTS

There have been no similar incidents of the PASS system causing another system to be outside of its design basis.

NRC FORM 366A
(4-95)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Fort Calhoun Station Unit No. 1	05000285	1998	- 009 -	01	7 OF 7

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

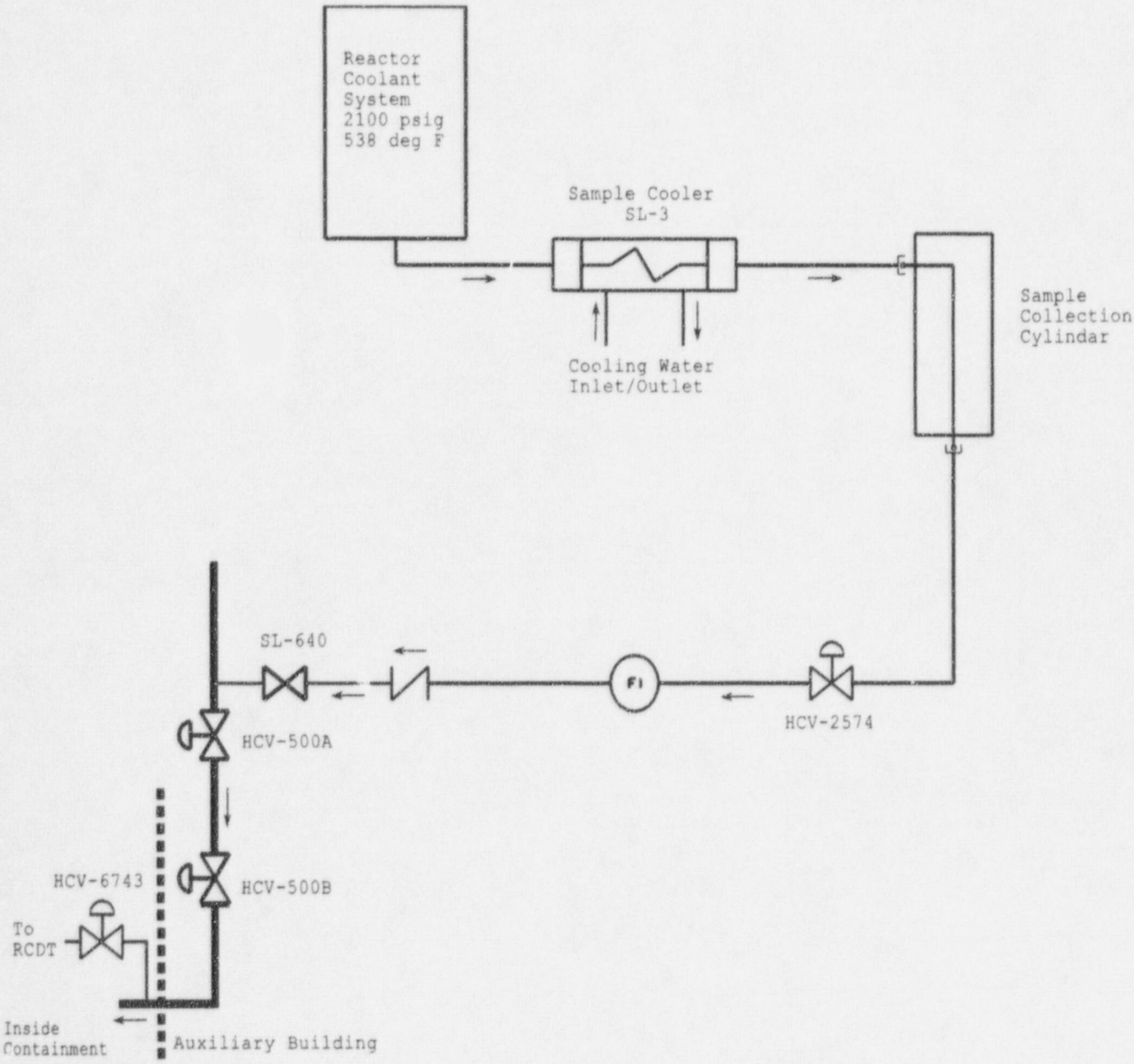

Figure 1