

NRC Form 366 (6-1998)						U.S. NUCLEAR REGULATORY COMMISSION						APPROVED BY OMB NO. 3150-0104    EXPIRES 06/30/2001  <small>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-4 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503</small>					
<b>LICENSEE EVENT REPORT (LER)</b>  (See reverse for required number of digits/characters for each block)																	
FACILITY NAME (1)  Cook Nuclear Plant Unit 1								DOCKET NUMBER (2)  05000-315				PAGE (3)  1 of 4					
TITLE (4)  Potential Failure Mode for Air Operated Components Not Considered in Original Design																	
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 DC Cook Unit 2		DOCKET NUMBER 05000-316						
11	28	1998	1998	-- 052 --	01	01	19	1999	FACILITY NAME		DOCKET NUMBER						
OPERATING MODE (9)		5		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)													
POWER LEVEL (10)		000		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)		<input checked="" type="checkbox"/> 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  Mr. Joel Gebbie, Safety Related Mechanical Engineering Supervisor								TELEPHONE NUMBER (Include Area Code)  616/465-5901 x1543									
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								
SUPPLEMENTAL REPORT EXPECTED (14)								EXPECTED SUBMISSION DATE (15)		MONTH	DAY	YEAR					
<input checked="" type="checkbox"/> YES	(If Yes, complete EXPECTED SUBMISSION DATE).						<input type="checkbox"/> NO			06	01	1999					
Abstract (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16) During a Safety System Functional Inspection (SSFI) self-assessment of the Auxiliary Feedwater (AFW) System, no analysis for a failure of the non-safety related remote speed controller manual loader for the Turbine Driven Auxiliary Feedwater Pumps (TDAFP) could be found. Engineering analysis of the condition resulted in the conclusion that a failure of the non-safety related manual loader output could result in the TDAFP operating at a minimum speed of 1900 rpm with only one Motor Driven Auxiliary Feedwater Pump (MDAFP) in operation when Auxiliary Feedwater (AFW) is required to mitigate the consequences of an accident. On November 28, 1998, this was determined to be reportable. Therefore, this LER is being submitted in accordance with 10 CFR 50.73 (a)(2)(ii)(B) as a condition outside the design basis of the plant. To prevent recurrence of this issue in the future, a training course will be provided to the Plant Engineering staff to ensure that the design requirements for non-safety related systems that interface with safety related systems are met. Potential modifications to the Auxiliary Feedwater System are being evaluated as an additional corrective action. When the course of action is decided, this LER will be updated to reflect that information. The safety significance of this condition was evaluated considering that the AFW system would be required to satisfy the accident analyses assumptions with the TDAFP operating at 1900 rpm in addition to a single active failure of a MDAFP. Under these conditions, the AFW system would not be able to produce AFW flow rates assumed in the accident analyses. However, the capability exists to cross-tie the AFW systems between units and provide an alternate source of AFW to the affected unit. Therefore, although this condition would result in the AFW system being outside the design bases, it would not pose a threat to the health and safety of the public. In December 1998 and January 1999 additional components were identified that could experience the same failure. After the investigations for those components are completed, a supplemental report to this LER will be submitted. That submittal is expected to occur by June 1, 1999.																	

9901260365 990119  
 PDR ADOCK 05000315  
 S PDR

LICENSEE EVENT REPORT (LER)  
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER(2)	LER NUMBER (6)				PAGE (3)	
		YEAR	SEQUENTIAL NUMBER		REVISION NUMBER		
		Cook Nuclear Plant Unit 1	05000-315	1998	--	052	--

Cook Nuclear Plant Unit 1

05000-315

2 of 4

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

**Conditions Prior to Event**

Unit 1 was in Mode 5, Cold Shutdown

Unit 2 was in Mode 5, Cold Shutdown

**Description of Event**

On September 21, 1998, a Safety System Function Inspection (SSFI) self-assessment of the Auxiliary Feedwater (AFW) system began, using SSFI techniques in accordance with NRC Inspection Procedure 93801, "Safety System Functional Inspection". The inspection team utilized a vertical slice review in the functional areas of engineering design and configuration control, operations, maintenance, surveillance and testing, and quality assurance and corrective actions. The self-assessment concluded on October 23, 1998.

During the SSFI, it was identified that no analysis for a failure of the non-safety related remote speed controller manual loader for the Turbine Driven Auxiliary Feedwater Pumps (TDAFP) existed. Engineering analysis of the condition resulted in the conclusion that a failure of the non-safety related manual loader output could result in the TDAFP operating at a minimum speed of 1900 rpm with only one Motor Driven Auxiliary Feedwater Pump (MDAFP) in operation when Auxiliary Feedwater (AFW) is required. On November 28, 1998, this was determined to be reportable in accordance with 10 CFR 50.73 (a)(2)(ii)(B) as a condition outside the design basis of the plant.

The steam supply to the Turbine Driven Auxiliary Feedwater Pump (TDAFP) is regulated by a Woodward governor valve. Depending upon the position of the governor valve, the speed of the TDAFP can vary from 1900 rpm to 4350 rpm. The governor valve is a pneumatically controlled device. The position of the governor valve may be remotely adjusted from the Control Room by using the speed control manual loader. The pneumatic input to the speed control manual loader is supplied from the 20 psig compressed air header. The loader output can vary from 3 psig to 15 psig. Loader output of 3 psig corresponds to maximum pump speed of 4350 rpm, while loader output of 15 psig corresponds to minimum pump speed of 1900 rpm.

Upon a loss of control air pressure to the loader, either due to loss of the compressed air system or a failure of the speed control manual loader, the TDAFP speed will automatically go to its design speed of 4350 rpm. This condition has been previously evaluated.

However, if the speed control manual loader fails such that the TDAFP governor valve receives the compressed air header pressure of 20 psig, the TDAFP speed will automatically go to its minimum speed setting of 1900 rpm. No evaluation of the failure mode where the TDAFP governor receives full compressed air system pressure of 20 psig had been performed.

The TDAFP speed control manual loader is a non-safety related component. As such, failure of the loader cannot be considered as the single active/passive failure of the AFW system. Therefore, the AFW system would have to satisfy the accident analyses assumptions with the TDAFP operating at 1900 rpm in addition to the single active failure of a MDAFP. Under these conditions, the AFW system would not be able to produce AFW flow rates assumed in the accident analyses.

In December 1998 it was identified that the Residual Heat Removal (RHR) heat exchanger air operated outlet valves would experience a similar failure to that identified for the TDAFP governor valve. In January 1999 it was identified that the Centrifugal Charging Pumps (CCP) discharge flow control valve was also susceptible to this failure mode. Information on these additional components will be provided after the investigations are completed.

**Cause of Event**

The root cause of this condition is that not all failure modes were considered when the compressed air system was originally designed. The plant was designed in accordance with General Design Criterion 26, "Protection Systems Fail-

LICENSEE EVENT REPORT (LER)  
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER(2)	LER NUMBER (6)				PAGE (3)
		YEAR	SEQUENTIAL NUMBER		REVISION NUMBER	
		1998	—	052	— 01	

Cook Nuclear Plant Unit 1

05000-315

3 of 4

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

Safe Design." The design criterion specifically refers to loss of air as a failure mode; however, it does not specify over-pressurization.

Review of components installed in the plant has revealed that a number of the components which utilize controllers are also designed with a dump solenoid. The dump solenoid ensures that the component (e.g., valve) will fail to its safe position. If a component was credited with an active safety function, it was fitted with a dump solenoid. However, if the component did not have an active safety function (such as the TDAFP speed controller manual loader), it was not designed with a dump solenoid.

**Analysis of Event**

This LER is submitted in accordance with 10 CFR 50.73(a)(2)(ii)(B) as a condition outside the design bases of the plant.

The AFW system provides water to the Steam Generators (SG) when main feedwater is unavailable because of a loss of main feedwater, unit trip, feedwater or steam line break, loss of off-site power, or small break Loss of Coolant Accident (SBLOCA). This water removes core residual heat to prevent the release of primary water through the pressurizer safety or power-operated relief valves and allows the plant to cool down to the point at which Residual Heat Removal (RHR) can be placed in service.

Each unit is equipped with one turbine driven AFW pump and two motor driven AFW pumps. For each unit, the TDAFP serves all four steam generators and each MDAFW pump serves two steam generators. The steam to the AFW pump turbine is supplied from two of the steam generators.

The preferred source of water for the AFW system is the non-safety related CST. Each unit's CST is cross-tied by a normally closed air operated valve to provide condensate to the opposite unit's AFW. If both CSTs are unavailable, water is supplied from Lake Michigan via the safety related ESW system, which is connected upstream of the AFW pump suction strainers. A minimum of 175,000 gallons is required to maintain the unit at hot shutdown for nine hours.

The TDAFP speed control manual loader is a non-safety related component. As such, failure of the loader cannot be considered as the single active/passive failure of the AFW system. Therefore, the AFW system would be required to satisfy the accident analyses assumptions with the TDAFP operating at 1900 rpm in addition to the single active failure of a MDAFP. Under these conditions, the AFW system would not be able to produce AFW flow rates assumed in the accident analyses. However, the capability exists to cross-tie the AFW systems between units. Procedures are in place and operators are trained to perform this evolution. Should this condition occur and since Cook is not designed for simultaneous design basis accidents in both units, supplemental AFW could be provided from the opposite unit. Therefore, although this condition would result in the AFW system being outside the design bases, it would not pose a threat to the health and safety of the public.

**Corrective Actions**

To ensure that this condition does not affect other components in the plant, a review has been performed to identify those components that utilize a similar controller. Several components with similar controllers have been identified. Those components are the CCP discharge flow control valve, the RHR heat exchanger outlet valves, and the SG PORVs (MRV-213, 223, 233, 243). The CCP and RHR valves were evaluated for the impact of the failure mode, and it was determined that corrective actions should be implemented. Additional information regarding corrective actions will be provided once the investigations for these components are completed. The PORVs were also evaluated and it was determined that the potential failure mode was adequately addressed by actions taken in the Emergency Operating Procedures.

This condition occurred because the effect of non-safety related components on the operation of safety related systems was not properly evaluated. To prevent recurrence of this condition, a training course will be provided to the Plant



LICENSEE EVENT REPORT (LER)  
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER(2)	LER NUMBER (6)				PAGE (3)
		YEAR	SEQUENTIAL NUMBER		REVISION NUMBER	
		1998	--	052	--	01

Cook Nuclear Plant Unit 1

05000-315

4 of 4

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

Engineering staff to ensure that the design requirements for non-safety related systems, structures or components (SSCs) that interface with safety related SSCs are met. This training will be completed prior to restart.

The current design change process has been verified to address requirements to design against single failures. The current procedure provides a significant improvement in the design change process over the process that was in place at the time of the original design for DC Cook

Potential modifications to the Auxiliary Feedwater System are being evaluated as an additional corrective action. When the course of action is decided, this LER will be updated to reflect that information.

**Previous Similar Events**

315/97-023-00

315/97-026-00

