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10CFR50.73

John L. Skolds
Vice President
Nuclear Operations

July 2, 1992

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

Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
LER 92-004, REVISION 1 (ONO 920039)

Attached is Licensee Event Report No. 92-004, Revision 1, for the Virgil C. Summer Nuclear Station. This revision changes the previously reported rating of the power supply cards from 1200 ohms to 1250 ohms and documents the completion of one of the additional corrective items. This report is submitted pursuant to the requirements of 10CFR50.73(a)(2)(iv).

Should there be any questions, please call us at your convenience.

Very truly yours,

John L. Skolds

CAC:nkk
Attachment

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IE22 1/1

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): Virgil C. Summer Nuclear Station

DOCKET NUMBER (2): 0 5 0 0 0 3 9 5

PAGE (3): 1 OF 0 6

TITLE (4): Reactor Trip Due To Low-Low Level In "C" Steam Generator

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 5	2 1	9 2	9 2	0 0 4	0 1	0 7	0 2	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 2 9

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)	X 50.73(a)(2)(iv)	73.71(b)
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 (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)
W. R. Higgins Supervisor, Licensing Support & Operating Experience	8 0 3 3 4 5 - 4 0 4 2

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
B	S0	FCV	F130	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 fifteen single-space typewritten lines) (16)

At approximately 0553 hours on May 21, 1992, the reactor tripped from 9% power on low-low steam generator water level in the "C" steam generator. Reactor power was being maintained at approximately 29% power when oscillations in feedwater flow, steam generator level, and steam flow occurred due to the automatic cycling of the steam line power operated relief valves (PORVs). The automatic cycling was caused by a malfunction of Bank 2 of the Steam Dump System, which resulted in an elevated reactor coolant system temperature and consequently elevated main steam line pressure. The cycling of the PORVs resulted in feedwater flow decreasing below the minimum 13% flow setpoint. When feedwater temperature decreased below 225°F, a feedwater isolation to "C" steam generator occurred. An attempt was made to reduce reactor power in order to transfer feedwater back to the emergency feedwater system. "C" steam generator level decreased below the low-low level reactor trip setpoint before this could be accomplished, and a reactor trip occurred.

The reactor protection system responded as designed. A post trip review revealed that six condenser steam dump valves did not fully open on demand. This was caused by replacement of "current to pneumatic" transducers that had higher impedance than the original transducers. The higher impedance prevented the power supply card from developing a full 20 milliamperes current, and the valves could not fully open on a 100% demand signal. The original transducers had been replaced under the "Equal To/Better Than" program.

The installed transducers were replaced with the original lower impedance transducers. The "Equal To/Better Than" procedure (Engineering Services - 419) is being reviewed to determine if it needs any program enhancements.

NRC Form 366
(9-83)

PLANT IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

EQUIPMENT IDENTIFICATION:

Flow Control Valve, EIIS System Code - SO
The higher impedance "current to pneumatic" transducers are Rosemount Mode 3311 Series transducers.

IDENTIFICATION OF EVENT:

In preparation for connecting the generator to the grid, with feedwater (FW) temperature at approximately 310°F and decreasing, reactor power was increased to approximately 29% in order to maintain FW flow above the minimum 13% FW isolation setpoint (FW isolation occurs when FW temperature is less than 225°F with FW flow less than 13%). While the turbine was rolling up, the main steam power operated relief valves (PORVs) cycled open and closed several times due to elevated main steam system pressure. The cycling of the PORVs resulted in FW flow decreasing below the minimum 13% flow setpoint. Before FW flow recovered to greater than the 17% low flow reset, FW temperature dropped below 225°F causing FW isolation to "C" SG. An attempt was made to reduce reactor power in order to transfer FW back to emergency feedwater system. "C" SG level dropped below the low-low SG level reactor trip setpoint and a reactor trip occurred from approximately 9% power.

EVENT DATE: May 21, 1992 at 0553 hours.

REPORT DATE: June 18, 1992

This report was initiated by Off-Normal Occurrence report 92-039.

CONDITIONS PRIOR TO THE EVENT:

Mode 1, 29% reactor power

DESCRIPTION OF EVENT:

On May 21, 1992, at approximately 0517 hours, Operations personnel began a main turbine startup. Reactor power was raised from 1% to approximately 15% power. At approximately 0521 hours, the main turbine began to roll up to 1800 revolutions per minute (RPM).

As the main turbine reached 900 RPM, FW temperature began to decrease at 10°F per minute. With 90 RPM acceleration rate selected, it would take another 10 minutes for the main turbine to reach 1800 RPM. However, in that time FW temperature would decrease from approximately 310°F to 210°F. With a FW isolation signal generated by a combination of low FW temperature (less than 225°F) and low FW flow (13%), a FW isolation signal became a concern. The actual time for the turbine to reach 1800 RPM took approximately 4.5 minutes longer than anticipated.

A decision was made to increase reactor power and dump steam to the condenser to raise feedwater flow above it's setpoint. Reactor power was raised to 29% power. The control room personnel received indication that Bank 1, steam dump valves, were fully open and Bank 2 was modulating open, which was expected for this bank. Bank 2, did not stop the reactor coolant system temperature rise. When this temperature reached approximately 574°F, main steam line pressure reached the main steam PORVs setpoint of 1148 pounds per square inch (psig).

The PORVs began to cycle open and closed in response to main steam line pressure. The PORVs cycling caused oscillations of steam flow, SG water level, and FW flow. FW flow on "C" FW line decreased to the 13% setpoint, while FW temperature was still above 225°F. However, before FW flow could increase above the bistable reset point of approximately 17%, FW temperature decreased to below 225°F and "C" FW isolation valve closed.

To preclude a reactor trip on low-low water level in "C" SG, an attempt was made to reduce reactor power to the point that the emergency feedwater system could sustain water level (approximately 3% reactor power). This was unsuccessful and the reactor tripped at 0553 hours on low-low level. With the exception of the previously mentioned problems, the plant response was normal. The plant was stabilized in Mode 3 until the cause of the malfunctions could be determined and corrected.

CAUSE OF EVENT:

The cause of the slow turbine roll up is unknown. Engineering and Instrument and Controls personnel performed troubleshooting on the turbine acceleration control circuitry. No abnormalities or equipment failures were found. During the subsequent main turbine startup and with 90 RPM acceleration rate selected, the turbine accelerated as expected with no anomaly in acceleration rate.

The cause of the steam dump malfunction was the inability of the power supply card to drive open the six Rosemount current to pneumatic (I/P) transducers. Under the "Equal To/Better Than" (ETBT) program, the Rosemount Model 3311 transducers were used to replace the Fisher Type 546 transducers in order to improve equipment reliability.

The Rosemount transducers were chosen due to their low tendency to drift and their stability. The Fisher Type 546 transducers had a high failure rate of both new and repaired transducers. Some Fisher transducers did not meet the required accuracy of 1% of the output range on initial installation. The ETBT evaluation compared the Fisher Type 546 and the Rosemount Model 3311 in function and performance in several areas. These were air consumption at 20 psig, output capacity at 20 psig, operating temperature limits, accuracy, humidity limits, remote output pressure reading, direct current input, air output, and vibration effects. In these areas, the transducers were compatible, so the evaluation concluded that the changing of transducers did not constitute a design change and the replacement was authorized. The post maintenance testing was a valve stroke using the transducer and a simulated input.

The Fisher transducers on steam dump valves were replaced with the Rosemount transducers between March 1, 1991 and April 18, 1991. On the steam dump valves to the condenser, two steam dump valves could be isolated by a single isolation valve, so their transducers were replaced and the valves were stroke tested.

The power supply cards for steam dump valve I/P transducers were rated to supply full load current (20 milliamperes) at an impedance of 1250 ohms and were bench tested to supply full load current up to 1900 ohms impedance. The impedance of the Fisher transducers is 176 ohms and of the Rosemount transducers a maximum of 410 ohms. With either two or three valves in steam dump banks 1, 3, and 4; the Rosemount transmitters would not exceed the power supply capacity. In Steam Dump Bank 2, however, there are 6 condenser valves. With the Rosemount transducers installed, the power supply card could not supply the necessary current to fully open the valves.

With the Bank 2 condenser dump valves not able to fully respond, reactor coolant temperature increased. This also increased main steam line steam pressure up to the main steam line PORVs setpoint resulting in the valves being activated.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Virgil C. Summer Nuclear Station	0 5 0 0 0 3 9 5	9 2	— 0 0 4	— 0 0	0 5 OF 0 6

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

ANALYSIS OF EVENT:

This report is being submitted pursuant to the requirements of 10CFR50.73(2)(a)(iv). Notification to the NRC Operations Center via the Emergency Notification System was made at 0845 hours on May 21, 1992, per the requirements of 10CFR50.72(b)(2)(ii).

With the exception of the steam dump operation and the turbine roll up, the plant responded as expected. The decreasing of reactor power was by control rod insertion. As previously stated, the faulty Bank 2 steam dump valves caused reactor coolant temperature and main steam line pressure to increase resulting in main steam line PORVs actuation. Because the PORVs were cycling open and closed: steam generator level, steam flow, and FW flow oscillations began to occur. FW flow decreased to 13% on "C" FW line and failed to increase above it's reset point of approximately 17% before FW temperature decreased below 225°F. This logic closed the "C" FW isolation valve. The reactor power was runback in an attempt to prevent a reactor trip on low-low steam generator water level. This was unsuccessful and the reactor tripped on low-low steam generator water level. The reactor coolant system cooled down to approximately 530°F during the transient and was stabilized in Mode 3.

IMMEDIATE CORRECTIVE ACTIONS:

Following the reactor trip, the operating shift placed the plant in a stable condition in accordance with Emergency Operating Procedure (EOP) 1.0, "Reactor Trip/Safety Injection Actuation," and EOP-1.1, "Reactor Trip Recovery." Operations personnel and the Independent Safety Engineering Group investigated the operation of the steam dump system. Instrument and Controls personnel performed tests of the I/P transducers to determine the response and loading of the Rosemount transducers. Once the cause of the steam dump system operation was identified, the Rosemount 3311 transducers on Bank 2, steam dumps, were replaced with the Fisher Type 546 transducers.

The turbine acceleration control circuitry was analyzed and the cause of the slow roll up was not discovered. The plant was authorized for a restart and turbine acceleration rate was monitored. It accelerated at 200 RPM with no anomalies.

A search of the other installed Rosemount transmitters confirmed that no more than three valves are driven by a single power supply card. Therefore the only such application of the Rosemount transducers in the plant was on the Bank 2 steam dumps.

ADDITIONAL CORRECTIVE ACTIONS:

The following additional actions were initiated by SCE&G to prevent a similar reoccurrence:

1. The "Equal To/Better Than" procedure (Engineering Services - 419) is currently undergoing an Engineering review to determine if this program needs any enhancement. This review will be complete by July 31, 1992. Any identified enhancements will be implemented on a schedule commensurate with their significance.

2. The calibration frequency of the Fisher Type 546 transducers that are on Bank 2 of the steam dump valves have been increased to counteract their tendency to drift.

PRIOR OCCURRENCES:

LER 84-009, dated March 5, 1984