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October 4, 2004
GO2-04-171

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397**
LICENSEE EVENT REPORT NO. 2004-004-00

Dear Sir or Madam:

Transmitted herewith is Licensee Event Report (LER) No. 2004-004-00 for the Columbia Generating Station. This report is submitted pursuant to 10 CFR 50.73(a)(2)(iv)(A). The enclosed report discusses items of reportability and corrective actions taken. Please note that this LER is not submitted within the 60-day time requirement of 10 CFR 50.73. The NRC Senior Resident Inspector at Columbia was notified of this on September 28, 2004 and a Condition Report has been initiated to document this issue in our corrective action program.

If you have any questions or require additional information, please contact Mr. DW Coleman at (509) 377-4342.

Respectfully,

RL Webring
Vice President, Nuclear Generation
Mail Drop PE04

Enclosure: Licensee Event Report 2004-004-00

cc: BS Mallett - NRC RIV
WA Macon - NRC-NRR
INPO Records Center
NRC Sr. Resident Inspector - 988C (2)
RN Sherman - BPA/1399
TC Poindexter - Winston & Strawn
WB Jones - NRC RIV/fax

IE22

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE

Reactor Scram Due to Failure of a Ceramic Capacitor on a NUCANA Servo Driver (NSD) Circuit Board

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
07	30	2004	2004 - 004 - 00			10	4	2004	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203 (a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Craig D. Sly	509-377-8616

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	JJ	CAP	AVX	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (If yes, complete 15. EXPECTED SUBMISSION DATE)	☒ NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On July 30, 2004, Columbia Generating Station (Columbia) was in Mode 1 with the reactor operating at 100 percent power. At 09:23 PDT, the reactor automatically scrammed when the reactor protection system (RPS) received trip signals from three out of four reactor steam dome pressure - high instrument channels.

The high reactor steam dome high-pressure condition was a result of a turbine governor valve (MS-V-GV/1) drifting closed. The turbine governor valve drifted closed due to a failure of a bypass capacitor on a NUCANA Servo Driver (NSD) circuit board associated with the governor valve electro-hydraulic control system. The failed capacitor was a monolithic ceramic capacitor. This capacitor provides high frequency bypass filtering for the onboard power supply at one of the operational amplifiers. The capacitor failed with low resistance which caused a high current load that eventually caused the circuit board protective fuse to clear and the closure of MS-V-GV/1.

This event posed no threat to the health and safety of the public. The NSD circuit board was replaced and a detailed failure analysis will be performed on the failed circuit board.

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

Event Description

On July 30, 2004, Columbia Generating Station (Columbia) was in Mode 1 with the reactor operating at approximately 100 percent of rated thermal power. At about 09:23 PDT, the reactor automatically scrammed when the reactor protection system (RPS) received trip signals from three out of four reactor steam dome pressure - high instrumentation channels.

Control room operators responded to the scram by implementing immediate operator actions as outlined in plant procedure PPM 3.3.1, Reactor Scram. An RPV low reactor water level signal (Level 3, water level less than 13 inches) was subsequently received which required entry into Emergency Operating Procedure PPM 5.1.1, RPV Control.

The initial actions of operators consisted of placing the mode switch in shutdown, monitoring reactor power (verifying average power range monitors (APRM's) were downscale), reactor pressure, and reactor water level. A control room operator then checked control rod position indication using the Rod Worth Minimizer and noted that all control rods indicated fully inserted except two. As required by PPM 3.3.1, the operator depressed all four manual scram pushbuttons and initiated the Alternate Rod Insertion (ARI) system. Operators then entered Emergency Operating Procedure PPM 5.1.2, RPV Control - ATWS. Per PPM 5.1.2 the Automatic Depressurization System (ADS) was inhibited and the High Pressure Core Spray pump (HPCS-P-1) was placed in manual. Within two minutes control room operators reported that all control rods were fully inserted and PPM 5.1.2 was exited.

At about 0953, the Shift Manager retrieved computer data indicating that an actual high reactor pressure condition that had exceeded the RPS setpoints had caused the scram. The Shift Manager reviewed the Emergency Action Levels (EALs) for the plant, obtained peer checks on his preliminary conclusions, and erroneously concluded that EAL 2.2.A.1 applied, requiring declaration of an Alert condition for the plant. EAL 2.2.A.1 states three requirements for declaring an Alert.

- Any RPS setpoint (including Manual) has been exceeded per TS 3.3.1.1, AND:
- RPS actuation failed to result in a control rod pattern which alone always assures reactor shutdown under all conditions, AND:
- Manual actions (mode switch in shutdown, manual pushbuttons, and ARI) result in reactor power less than or equal to 5 percent.

At 10:00 PDT an Alert was declared based on an RPS setpoint having been exceeded (reactor pressure-high) and the erroneous conclusion that two control rods initially did not fully insert following the automatic reactor scram. The Emergency Response Organization was notified by pager and responded. At about 10:37 and 10:48 PDT, the HPCS and ADS systems were restored to their normal lineups. At about 11:57 PDT, the Alert was terminated because all control rods were known to be inserted into the core and all required emergency systems were operable. The Alert declaration was subsequently retracted on August 3, 2003. A review of control rod position indication from the Plant Data Information System (PDIS), Rod Worth Minimizer logs, and Automatic Scram Timer data showed that all control rods were successfully inserted to the "Full-In" position following the initial RPS actuation. This information was available at 10:00 PDT but had not been reviewed when the decision to enter EAL 2.2.A.1 was made.

After the reactor tripped on high reactor pressure, the turbine bypass valves opened and reduced reactor pressure. Reactor water level increased to above Level 8 (high water level, greater than 54.5 inches), which caused both reactor feedwater pump turbines to trip. Reactor water level then decreased to approximately 20 inches at about 09:38 PDT. A Control Room operator attempted to restart reactor feedwater pump (RFW-P-1A) but its turbine (RFW-DT-1A) would not reset. The operator then successfully reset reactor feedwater pump turbine (RFW-DT-1B) and used reactor feedwater pump RFW-P-1B to raise water level to about 35 inches with reactor pressure slowly decreasing.

Cause of Event

The physical cause of the reactor scram was an actual high reactor steam dome pressure condition that resulted from closure of a turbine governor valve (MS-V-GV/1). The turbine governor valve closed due to a failure of a NUCANA Servo Driver (NSD) (DEH-EC-CIK25/NSD) circuit board in the turbine digital electronic control system (DEH). The NSD circuit board failure caused governor valve MS-V-GV/1 to initially fully open and then drift closed. As the governor valve closed, it restricted steam flow from the reactor vessel and caused reactor vessel pressure to increase to the RPS high-pressure trip setpoint.

The NSD circuit board is a position controller that interfaces with an Electro-Hydraulic (EH) actuator. Actuator position is determined by a circuit that interfaces with a linear variable differential transformer. Input to the circuit is the summation of the governor valve demand, valve test, governor valve close bias, and governor valve optimization bias.

The component that failed on the NSD circuit board was a monolithic ceramic capacitor. This specific capacitor provided high frequency bypass filtering for the onboard power supply at one of the operational amplifiers. The capacitor failed with low resistance causing a high current load that eventually caused the NSD circuit board protective fuse to clear.

Ceramic capacitors are a high reliability electronic component and are extensively used in electronic circuits. Ceramic capacitors can fail either open or short (low resistance) and the failure is normally catastrophic and not predictable. Each NSD circuit board has 36 monolithic ceramic capacitors and a total of 56 ceramic capacitors. There are 12 NSD circuit boards in the turbine DEH system and many other circuit boards in the DEH system that have a similar quantity of ceramic capacitors. All 12 NSD circuit boards have been in service since initial plant startup.

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Based on information contained in Military Handbook 217F(MIL-HNBK-217F, Reliability Prediction of Electronic Equipment, dated December 2, 1991) it has been concluded that the failure rate of ceramic capacitors is low and the likelihood of a similar failure during the remaining life of the plant is considered low.

Safety Significance

A reactor vessel pressure increase during reactor operation compresses the steam voids in the core and results in increased reactivity. This causes increased core heat generation that could lead to fuel failure and system over-pressurization. The purpose of the reactor high-pressure scram function is to counteract a pressure increase by quickly reducing core fission heat generation. No specific safety analysis in the Final Safety Analysis Report takes credit for the high-pressure reactor scram function. The high-pressure reactor scram works in conjunction with the pressure relief system to prevent reactor vessel pressure from exceeding the maximum allowable pressure.

This event posed no threat to the health and safety of the public. Reactor vessel pressure did not exceed design or code limits during the event and no safety relief valves operated. Plant safety systems responded as expected with three exceptions:

1. Two control rods did not immediately indicate fully inserted on the Rod Worth Minimizer. Within two minutes, the Rod Worth Minimizer indicated that these two control rods were fully inserted into the core. A subsequent review of control rod position indication from the Plant Data Information System (PDIS), Rod Worth Minimizer logs, and Automatic Scram Timer data showed that all control rods were successfully inserted to the full-in position following the initial RPS actuation. A Condition Report was written to document that two control rods did not initially indicate full-in. A computer generated "All Rod's In" display was developed for the Plant Process Computer Replacement System (PPCRS) to assist Control Room operators in assessing if all rods are inserted after a reactor scram. This new display mode will enhance Control Room operators' ability to make timely and accurate evaluations of EAL's involving reactivity control in the future.

2. Following the reactor scram, a wetwell-to-drywell vacuum breaker CVB-V-1CD rear disc was discovered to be indicating open during a panel walkdown performed by operators in the control room. A Condition Report was written to document that the rear disk to this vacuum breaker indicated open after the scram. Troubleshooting of the indication circuitry determined the disc was actually closed. A relay in the position indication circuit for the valve was replaced and position indication was restored. A drywell-to wetwell differential pressure did not exist during the event that would cause a CVB to open. The failed relay is not safety related and not required to support the safety function of the vacuum breaker. A Problem Evaluation Request had been initiated on April 27, 2004 documenting that historically Columbia has had a higher than desired frequency of indication problems with containment vacuum breakers. Broader corrective actions to address vacuum breaker indication problems are currently planned under this PER.

3. Operators attempted to reset reactor feedwater pump turbine (RFW-DT-1A) and it would not reset. Operators then reset reactor feedwater pump turbine RFW-DT-1B successfully and used reactor feedwater pump RFW-P-1B to raise reactor water level. A Condition Report was written to document that RFW-DT-1A did not reset after tripping on high reactor level. It was later determined that the cause of the failure of RFW-DT-1A to reset was that the feedwater pump turbine governor valve position did not indicate fully closed (indicated 6 percent open) which prevented the feedwater pump turbine from resetting. The cause was a failure of a Magton actuator associated with the turbine control oil system to achieve minimum pressure after the turbine initially tripped. The Magton actuators for both reactor feedwater pumps were disassembled, cleaned, lubricated and reinstalled. They were then post-maintenance tested.

This event is reportable in accordance with 10 CFR 50.73 (a)(2)(iv)(A), any event or condition that resulted in manual or automatic actuation of the reactor protection system (RPS) including reactor scram.

Immediate Corrective Actions

1. The failed NSD circuit board was replaced and post-maintenance testing was completed. A Condition Report was written to document the NSD circuit board failure.
2. A root cause analysis was initiated to investigate the causes and extent of condition for the NSD circuit card failure.
3. A calibration check was performed on reactor steam dome pressure-high instrument channel MS-PS-23B, which did not trip during the event. This instrument channel was found set within its setpoint tolerance.

Further Corrective Actions

1. The failed circuit board will be sent offsite for a detailed failure analysis. Additional corrective actions will be taken, as deemed necessary, based on the results of the failure analysis.
2. Appropriate actions have been taken to address station performance during the emergency declaration and response.

Previous Similar Events

There have been no previous similar events involving turbine digital electronic control system (DEH) NSD card failures at Columbia Generating Station.

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EIIS Information

Text Reference	System	Component
Bypass Capacitor	JJ	CAP
Circuit Board	JJ	ECBD
Turbine Governor Valve	JJ	FCV
Turbine Digital Electronic Control System	JJ	N/A
High Pressure Core Spray System	BG	N/A
Automatic Depressurization System	BM	N/A
Reactor Protection System	JC	N/A
APRM	IG	N/A
ARI	JC	N/A
Reactor Feedwater Pump	SJ	P
Vacuum Breaker	BF	VACB