ENERGY NORTHWEST

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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. 2009-004-01

Dear Sir or Madam:

This submittal provides an update to Licensee Event Report No. 2009-004-00, which reported a reactor scram event at Columbia Generating Station on August 5, 2009. This revision adds contributing causes for the event based on a recent re-evaluation of the event.

There are no commitments being made to the NRC herein. If you have any questions or require additional information, please contact Ms. L. L. Williams at (509) 377-8418.

Respectfully,

B. J. Sawatzke
Vice President, Nuclear Generation & Chief Nuclear Officer

Enclosure: Licensee Event Report 2009-004-01

cc: NRC Region IV Administrator
NRC NRR Project Manager
NRC Senior Resident Inspector/988C
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NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 10/31/2013
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: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Section (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resource@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.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Columbia Generating Station	05000397	1 OF 4

4. TITLE
6.9 kV Non-Segregated Electrical Bus Failure

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
08	05	2009	2009	004	01	05	02	2011	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
1	☐ 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)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
100	☐ 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
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
Michael A. Huiatt, Principal Licensing Engineer	509-377-4243

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
E	EA	NSBU	DE36	N					

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)*

At 0750 on August 5, 2009, while Columbia was operating at 100% power, an electrical fault on a 6.9 kilovolt (kV) non-segregated bus resulted in a turbine trip and automatic reactor scram. During the scram recovery, the turbine bypass valves remained in the full open position and did not automatically modulate to maintain reactor pressure as expected.

The most probably cause of the bus failure is a relaxation of bolted connections on the center phase flexible link(s) caused by repeated thermal cycles over time. The root cause identified for this event was the non-performance of preventive maintenance (PM) tasks for torque checks of the non-segregated bus links. The damaged bus has been repaired, and changes to the controls for PMs will be made to ensure proper performance to prevent this type of failure from recurring.

The failure of the bypass valves to control pressure resulted from an error introduced during a design change of the Digital Electric Hydraulic (DEH) system implemented in 2007. As a result, the bypass valves transferred to manual pressure control mode when the pressure transient exceeded the main steam throttle inlet pressure input signal limit for all three channels. The DEH system has been modified to correct the erroneous signal limit, and to retain automatic control of the bypass valves during similar high throttle pressure events.

No previous similar events have been reported by Columbia.

NARRATIVE

Plant Condition

The plant was in Mode 1 operating at 100% when this event occurred.

Event Description

On August 5, 2009, at approximately 0750 the plant experienced an automatic actuation of the Reactor Protection System [JC] due to a main turbine trip. The turbine trip occurred immediately following a main generator [EL] differential lockout due to an electrical fault in the 6.9 kV electrical distribution system [EA]. The electrical fault was determined to have been caused by a failure in the non-segregated electrical bus [NSBU] which generated enough smoke in the turbine building to require declaration of an unusual event at 0812 due to toxic gases in amounts that could affect the health of plant personnel or safe plant operation.

The plant response to the disturbance in the electrical system and subsequent turbine trip and scram (RPS actuation) was as expected with the following exceptions:

1) The DEH system [JI] unexpectedly transferred to manual pressure control mode with the bypass valves remaining in the full open position. Consequently, reactor pressure dropped from a post turbine trip maximum of approximately 1081 psig to an approximate minimum of 396 psig in just over 3.5 minutes. Plant Operators terminated the reactor pressure decrease by manually closing the inboard Main Steam Isolation Valves.

2) After the scram, the reactor coolant temperature dropped 106 degrees over a period of approximately six minutes, exceeding the Technical Specification (TS) allowable cool down rate of 100 degrees in an hour.

3) The feedwater [SJ] pumps tripped on low suction pressure within a few seconds of the scram. Water from the feed system was re-established when reactor pressure decreased to less than the shutoff head of the condensate booster pumps approximately two minutes after the scram.

There was no inoperable equipment at the start of the event that contributed to the event. This LER is submitted pursuant to 50.73(a)(2)(iv)(A) as an event or condition that resulted in automatic actuation of the Reactor Protection System.

Causes

The most probable cause of the electrical bus failure was a loosening or relaxation of bolted connections on the center phase flexible link(s) due to repeated thermal cycles over time. The looseness resulted in increased heating of the joint, which degraded the joint. It is postulated that heating occurred in the joint to the point that insulation degraded, resulting in the formation of a high energy arc fault which shorted between phase conductors. The loose connections were not detected prior to the event due to a failure to perform PMs for bus torque checks as well as a lack of adequate temperature monitoring of the non-segregated buses. Because the bus was destroyed, little forensic evidence exists to support these conclusions.

NARRATIVE

Subsequently, it was discovered that the bus had been improperly uprated in 1994 to support a modification that added the adjustable speed drive (ASD) loads to the bus. The increased bus loading resulted in reduced margin between the bus loading and the actual bus rating, leading to increased bus link temperatures. In addition, it was discovered that the ASD modification resulted in the addition of non-linear loads to the bus. This modification was installed in 1996. The non-linear loads caused an increase in total harmonic distortion (THD), which also results in increased heating on the bus. The impact of this increase in THD was not addressed as part of the design change. In both instances, less than adequate engineering rigor was used to analyze the impact of a plant design change on the electrical bus. As such, this lack of engineering rigor has been identified as a contributing cause to the electrical bus failure.

The unexpected transfer of the DEH system to manual pressure control was due to a design error introduced during the DEH system modification that was installed in 2007. The failure threshold in the DEH software code for the main steam throttle inlet pressure transmitters was set too low to accommodate the pressure spike that occurred when the turbine tripped from full power.

Design changes that were made to the feedpump suction trip setpoints and the digital feedwater reactor level control system in R18 (2007) contributed to the feedpumps tripping on low suction pressure. Following the scram, the feedpump turbines ramped up speed to respond to the RPV low level signal, which caused an unanticipated drop in suction pressure below the trip limit. The change that staggered the feedpump suction trip setpoints did not achieve the desired result of preventing both feedpumps from tripping off in this event.

<u>Corrective Actions Taken or Planned</u>

The following actions have been taken or are planned to address the issues identified above, prevent recurrence, and address the extent of condition/cause:

1) The damage to the 6.9 kV bus has been repaired, and windows have been installed in the bus duct covers to allow for more direct thermography monitoring of the links.

2) An appropriate frequency for thermography checks will be established in conjunction with performing torque checks for the other non-segregated buses during the next refueling outage to prevent recurrence of this type of event. The content and controls for PMs will be strengthened to ensure proper completion of critical steps, and appropriate levels of review and approval are applied to any changes.

3) The DEH system software has been modified to increase the failure threshold setpoint for the main steam throttle inlet pressure signal and to ensure the system will remain in automatic pressure control upon failure high of the main steam throttle inlet pressure signal.

4) Because the TS allowed cool down rate of the reactor pressure vessel (RPV) was exceeded, a fatigue analysis was performed on the RPV and concluded that this event is bounded by transients evaluated in the original RPV fatigue analysis.

NARRATIVE

5) The other non-segregated buses were visually inspected and had insulators torque checked. Additional infrared thermography windows were installed in the bus housing covers at all flexible and rigid link locations with the exception of the two buses that are only normally energized during startups up to 23% power, in which only some of the thermography windows were installed. The remainder of the windows for the two startup buses are scheduled for installation during the next refueling outage in 2011.

6) The logic for the Digital Feedwater level control system has been modified to limit feedwater pump turbine speed in response to a reactor scram. A staggered time delay for feedwater pump low suction pressure trips has also been added to allow suction pressure to recover before a second pump trips.

7) The non-segregated electrical bus is being upgraded to replace the existing busbar with a larger busbar with welded bus connections to increase the bus ampacity rating and regain margin between the bus loading and bus rating.

Assessment of Safety Consequences

For this event, all Emergency Core Cooling Systems (ECCS) were available to perform their intended safety functions. Both off-site power circuits were available and all three of the emergency diesel generators [EB and EK] were operable and available. This event did not involve an event or condition that could have prevented the fulfillment of any safety function described in 10 CFR 50.73(a)(2)(v). Therefore, this event posed no threat to the health and safety of the public or plant personnel and was of low safety significance.

Similar Events

No previous similar bus failures have been reported by Columbia. A review of the Corrective Action Program condition report database found one other occurrence of an event with similar characteristics. In May 2007 evidence of electrical connection heating was discovered during performance of a PM on one of the non-segregated buses (not the bus that failed in this event). The connection was reworked and re-taped. Visual inspections conducted to address the extent of condition on the other non-segregated buses identified and repaired instances of potential overheating on one bus (not the bus that failed in this event) and identified no major concerns for the bus that failed in this event.

Energy Industry Identification System (EIIS) Information

EIIS codes are bracketed [ ] where applicable in the narrative.