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January 18, 2005
GO2-05-011

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. 2004-008-00

Dear Sir or Madam:

Transmitted herewith is Licensee Event Report No. 2004-008-00 for the Columbia Generating Station. This report is submitted pursuant to 10 CFR 50.73(a)(2)(v)(D). The enclosed report discusses items of reportability and corrective actions taken.

If you have any questions or require additional information, please contact Mr. GV Cullen at (509) 377-6105.

Respectfully,

WS Oxenford
Vice President, Nuclear Generation
Mail Drop PE04

Enclosure: Licensee Event Report 2004-008-00

cc: BS Mallett – NRC RIV
BJ Benney – NRC-NRR
INPO Records Center
NRC Sr. Resident Inspector – 988C (2)
RN Sherman – BPA/1399
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IE22

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2007
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 Core Isolation Cooling Isolation Due to Inadvertant Closure of Containment Isolation Valve

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
11	22	04	2004-008-00			01	18	05	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
☐ 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

NAME	TELEPHONE NUMBER *(Include Area Code)*
Craig Sly – Principal Engineer, Licensing	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

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 November 22, 2004, Columbia Generating Station (Columbia) was in Mode 1 at approximately 100 percent of rated thermal power. At 17:30 PST, the Reactor Core Isolation Cooling (RCIC) system was declared inoperable after one of its steam supply containment isolation valves (RCIC-V-63) was inadvertently closed during the performance of a channel functional test/channel calibration procedure. The procedure was discontinued and plant operators verified that the High Pressure Core Spray System was operable as required by Technical Specifications. The RCIC system was restored to its normal standby lineup and declared operable two hours and three minutes later.

The immediate cause was a personnel error by one of the I&C technicians performing the procedure. The root causes included over-reliance on self-checking and peer-checking, the procedure did not contain adequate precautionary information, no direct field supervision of the evolution, and no integrated risk assessment of the work.

A briefing was conducted to explain the event to Electrical and I&C craft and supervisors. The procedure being used, and other similar procedures, will be revised to add appropriate precautions. Additional expectations for frequency, depth, and quality of supervisory field oversight are being implemented. A new integrated risk management procedure will reduce the potential for similar errors.

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

I. Event Description

On November 22, 2004 Columbia Generating Station (Columbia) was in Mode 1 with the reactor operating at approximately 100 percent of rated thermal power. At approximately 17:30 PST, the Reactor Core Isolation Cooling System (RCIC) steam supply line outboard containment isolation valve (RCIC-V-63) closed. This caused RCIC to become inoperable.

The closure of RCIC-V-63 occurred while performing procedure ICP-RCIC-Q901, "RCIC Isolation on RCIC Steam Flow High Div 2 - CFT/CC" (Channel Functional Test/Channel Calibration). Procedure ICP-RCIC-Q901 provides instructions for the channel functional test and channel calibration of the RCIC steam line flow high instrument channel associated with RCIC-DPIS-7B.

Two technicians were setting up to perform a channel functional test/channel calibration of differential pressure indicating switch RCIC-DPIS-7B. RCIC-DPIS-7B senses steam flow in the steam supply piping to the RCIC turbine. The switch contacts on RCIC-DPIS-7B close on high steam flow conditions (normally open contacts) to make up RCIC-V-63 isolation logic, close RCIC-V-63, and activate an annunciator in the control room.

The procedure calls for isolating RCIC-DPIS-7B, connecting water pots to its test valves, and connecting a pressure source and test gauge to the high pressure side water pot. Then the procedure calls for connecting a digital multi-meter (DMM) across terminals AA1 and AA2. The purpose of the DMM is to indicate when the high steam relay flow contacts close during the test. Once the pressure source is connected to the high pressure side water pot, and the DMM is connected, the procedure calls for opening the breaker to the RCIC-V-63 motor operator so the valve will not close when RCIC-DPIS-7B is actuated during the test.

The DMM that was connected across terminals AA1 and AA2 had been taken to the instrument rack with its test leads stacked. That is, the banana jack leads were connected to each other (stacked) at the negative terminal of the DMM. This is typical of how DMM's are transported by the technicians. The technicians connected the leads to terminals AA1 and AA2 without un-stacking the leads at the DMM. This effectively jumpered out the contact, actuated RCIC-DPIS-7B, closed RCIC-V-63 and caused an alarm in the control room.

Control room operators declared RCIC inoperable, entered the appropriate Technical Specification Required Actions, and directed that technicians back out of the procedure.

II. Cause of Event

The immediate cause of this event was a personnel error by one of the I&C technicians performing the procedure. The technician who connected the DMM across contacts AA1 and AA2 of RCIC-DPIS-7B failed to use proper self-checking techniques to ensure that the DMM was properly configured before connecting the DMM across the contacts. This effectively jumpered out the contact the technicians were attempting to monitor during performance of the procedure and resulted in an actuation of RCIC-DPIS-7B and isolation of RCIC-V-63.

A root cause analysis was performed regarding this event. The root cause analysis determined the following:

- There was an over-reliance on self-checking and peer-checking during performance of the procedure.
- The procedure did not contain precautionary information associated with performing steps in the specified sequence or the potential for a single human error to cause a safety system isolation.
- There was no direct field supervision of the work.
- There was no pre-work review to identify the potential risk or consequences.

III. Safety Significance

The RCIC system is designed to operate either manually or automatically following reactor pressure vessel (RPV) isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of RPV water level. Under these conditions, the High Pressure Core Spray (HPCS) and RCIC systems perform similar functions. After the loss of RCIC, control room operators entered Technical Specification (TS) Action 3.5.3.A, which requires that with the RCIC system inoperable, the HPCS system must be verified operable immediately and the RCIC system must be restored to an operable status within 14 days. Since HPCS was verified to be operable and the total duration of this event was approximately two hours and three minutes, the TS requirements were satisfied. Therefore, this event posed no threat to the health and safety of the public or plant personnel.

This event is reportable under 10 CFR 50.73 (a)(2)(v)(D), "Any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident." There were no structures, systems or components that were inoperable at the start of the event that contributed to the event.

IV. Immediate Corrective Actions

Control room operators entered and complied with TS Action Statement 3.5.3.A, by verifying that HPCS was operable. Operators then proceeded to take action to protect HPCS. Control room operators directed the I&C technicians to back-out of the procedure and restore all components to their normal line-up. After all components were restored to their normal line-up, RCIC was placed in a normal line-up, with RCIC-V-63 open, and declared operable two hours and three minutes after the isolation occurred.

V. Further Corrective Actions

A stand-down briefing was conducted on the day after the event to explain the event to Electrical and I&C craft and supervisors.

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ICP-RCIC-Q901 and other similar procedures will be revised to either:

- Reduce the consequences of a single human error by changing the status of the system or equipment such that it cannot be inadvertently actuated or isolated while hooking up test equipment, or;
- Add appropriate cautions which will enhance worker awareness regarding risk, consequence, and mitigating actions.

Formal expectations for functional testing of test equipment and self-checking and peer-checking of test equipment set-up prior to connecting to in-service systems will be added to management and job observation forms.

Specific expectations for depth and quality of supervisory field oversight for medium and high risk activities will be added to existing expectations.

Plant Procedure Manual (PPM) 1.3.76, Integrated Risk Management, is a relatively new procedure at Columbia. This procedure establishes administrative controls for the oversight of risk significant activities in Modes 1, 2, and 3. PPM 1.3.76 has been in a pilot period and utilized only for a select number of work activities. This procedure was not used for the performance of ICP-RCIC-Q901 on November 22, 2004. This procedure is currently in the process of being fully implemented.

VI. Previous Similar Events

A search of the Columbia Problem Evaluation Request (PER) database and LER database was conducted. The search covered the period from 1999 to the date of this event and identified one similar event involving an RCIC-V-63 isolation due to personnel error. The event is documented in LER 397-2003-008. A Condition Report was initiated (2-04-06501) to document that the corrective actions associated with LER 397-2003-008 were not effective in preventing the recurrence of this similar event. The corrective actions listed above were developed to minimize recurrence of similar events in the future.

VII. EIIS Information

Text Reference	System	Component
Reactor Core Isolation Cooling System	BN	
High Pressure Core Spray System	BG	
RCIC-DPIS-7B	BN	PDIS
RCIC-V-63	BN	ISV