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January 2, 2007
GO2-07-002

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. 2006-002-00**

References: 1) Letter dated November 7, 2005, C.E. Johnson (NRC) to J.V. Parrish (EN), "Columbia Generating Station – NRC Integrated Inspection Report 05000397/2005004."

Dear Sir or Madam:

Transmitted herewith is Licensee Event Report No. 2006-002-00 for Columbia Generating Station. This report is submitted pursuant to 10 CFR 50.73(a)(2)(v)(B) as specified by NRC in Integrated Inspection Report 05000397/2005004 (Reference 1). The enclosed report discusses items of reportability and corrective actions taken.

There are no new regulatory commitments being made. If you have any questions or require additional information, please contact Mr. GV Cullen at (509) 377-6105.

Respectfully,

D. K. Atkinson

DK Atkinson
Vice President, Nuclear Generation
Mail Drop PE08

Enclosure: Licensee Event Report 2006-002-00

cc: BS Mallett – NRC RIV
RF Kuntz – NRC NRR
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NRC Sr. Resident Inspector – 988C (2)
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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 information 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

Shutdown Cooling Isolation due to Inadequate Procedure Step

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 0500
11	03	2006	2006 - 002 - 00			01	02	2007	FACILITY NAME	DOCKET NUMBER 0500

9. OPERATING MODE
4

10. POWER LEVEL
00

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

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
Stephen Mazurkiewicz, Senior Licensing Engineer	509-377-8463

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 3, 2006 at 0309 PST, Shutdown Cooling (SDC) was inadvertently removed from service while in Mode 4. The loss of SDC resulted from the isolation of the inboard primary containment isolation valve (RHR-V-9) on the Residual Heat Removal (RHR) SDC common suction header.

The isolation occurred while transferring Reactor Protection System (RPS) B to its alternate power supply. During the transfer electrical disconnect RHR-DISC-V/9 was opened per Plant Procedures Manual (PPM) 2.7.6 to prevent a loss of SDC. Use of RHR-DISC-V/9 during the transfer resulted in an unintended containment isolation signal to RHR-V-9. The cause of this event was an inadequate procedure step derived from inaccurate technical information in procedure SOP-RHR-SDC-BYPASS.

Corrective action has been initiated to revise subject procedures. More specific guidance for validation will be provided as part of the Procedure Review Program and expectations for technical accuracy will be reinforced.

There have been two reported isolations of SDC at Columbia within the past 5 years (LER 2003-003 and LER 2003-005).

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

Plant Conditions

At the time of the event, Columbia Generating Station (Columbia) was in Mode 4 (cold shutdown) with reactor vessel level [AD-RPV] at 70 inches and reactor coolant temperature at 114 degrees Fahrenheit. There were no structures, systems, or components (SSCs) inoperable at the start of the event that contributed to the event.

Event Description

On November 3, 2006, at approximately 0309 PST, an isolation of the Residual Heat Removal (RHR) Shutdown Cooling (SDC) [BO] common suction header occurred when the inboard primary containment isolation valve (RHR-V-9) [ISV] closed. The RHR SDC isolation occurred while performing PPM 2.7.6, "Reactor Protection System," Section 5.9, Shifting Reactor Protection System (RPS) [JC] B to ALT B or Normal Power Supply. During the RPS B transfer, electrical disconnect RHR-DISC-V/9 [DISC] was opened, per Step 5.9.6 of Plant Procedures Manual (PPM) 2.7.6, to disable RHR-V-9 and maintain SDC. Step 5.9.17 of PPM 2.7.6 required that disconnect RHR-DISC-V/9 be closed after the RPS B transfer was complete. Contrary to the intent of the procedure, opening disconnect RHR-DISC-V/9 did not remove control power from the RHR-V-9 isolation logic and created a sealed-in close signal during the transfer that isolated SDC when power was restored to RHR-V-9.

At the time of the event, RHR SDC subsystem B was operating in the SDC mode and RHR SDC subsystem A was available for SDC service but not in operation. Reactor Recirculation Pump 1A (RRC-P-1A) [AD-P] was running to support reactor core circulation and was unaffected by the RHR SDC isolation. Alternate means of decay heat removal were available at the time of the event.

The SDC isolation was caused by closure of RHR-V-9, the inboard primary containment isolation valve in the common suction line for both RHR SDC subsystems. Closure of RHR-V-9 subsequently tripped RHR pump 2B (RHR-P-2B). Operators received the "RHR B Pump Trip" alarm and entered abnormal condition procedure ABN-RHR-SDC-LOSS, "Loss of Shutdown Cooling," and ABN-LEVEL, "Unplanned Water Level Change."

Operators reopened RHR-V-9 and restored SDC with RHR SDC subsystem B at 0355 PST. During the time that RHR SDC was out of service, reactor coolant temperature increased to 148 degrees Fahrenheit and vessel level reached 95 inches. Reactor temperature and level were restored to their previous operating bands by 0451 PST.

Immediate Corrective Action

Prior to the event, operators had recently been briefed on a loss of SDC and were prepared to respond.

Control Room Operators entered ABN-RHR-SDC-LOSS and ABN-LEVEL to manage the event. PPM 2.7.6 was reviewed to determine what step was being performed at the time the SDC isolation occurred. Valve travel to the fully closed position removed the sealed-in close isolation signal and RHR SDC subsystem B was placed back into service.

The applicable drawings were reviewed and confirmed that the actuations/isolations that occurred should have been expected.

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Cause of Event

The cause of this event was an inadequate procedure step in PPM 2.7.6 that resulted in an unintended and undetected sealed-in isolation signal to RHR-V-9. RHR-V-9 has two disconnects, RHR-42-8BA2A and RHR-DISC-V/9. Power for RHR-V-9 control logic is provided through disconnect RHR-42-8BA2A. Using disconnect RHR-DISC-V/9 to maintain SDC during the RPS transfer created a sealed-in isolation signal to RHR-V-9 that isolated SDC upon restoration of power to the valve.

The unintended action is the result of inaccurate technical information incorporated into PPM 2.7.6 from another approved procedure. Using RHR-DISC-V/9 to maintain SDC during RPS B transfers was originally established in SOP-RHR-SDC-BYPASS, "Bypassing RHR Shutdown Cooling Isolation Logic in Mode 4 and 5," Revision 0 dated August 10, 2004 and later incorporated in PPM 2.7.6, Revision 20 dated May 30, 2005.

Contrary to the existing expectation in SWP-PRO-02, "Preparation, Review, Approval, and Distribution of Procedures," on responsibilities for technical adequacy, the adverse impact of originally designating disconnect RHR-DISC-V/9 rather than disconnect RHR-42-8BA2A in SOP-RHR-SDC-BYPASS was not recognized. Therefore, a contributing cause of this event is inadequate scope within the Procedure Review Program as the guidelines in SWP-PRO-02 for procedure validation lack clear direction.

Assessment of Safety Consequences

This event did not pose a threat to the health and safety of the public or plant personnel.

The RHR SDC was out of service for approximately 46 minutes during which the reactor level increased from 70 inches to 95 inches and the reactor coolant temperature increased from 114 degrees Fahrenheit to 148 degrees Fahrenheit. The increase in reactor level is partially attributed to the isolation of the previously established letdown path through RHR subsystem B. The time-to-boil was calculated to be 2.3 hours as of 0309 on 11/3 at the onset of the isolation.

At the time of the event, RRC-P-1A was running to support reactor core circulation and was unaffected by the RHR SDC isolation. The drywell was purged and High Pressure Core Spray (HPCS) [BG] and Condensate System [SD] were unavailable. Technical Specification 3.4.10, "RHR Shutdown Cooling System – Cold Shutdown," allows both RHR SDC subsystems and recirculation pumps to be shutdown for a period of 2 hours in an 8 hour period on the basis that core heat generation is low enough and the associated heatup rate slow enough that RHR flow interruptions can be tolerated.

Alternate decay heat removal was available by venting to the suppression pool and ABN-RHR-SDC-LOSS, "Loss of Shutdown Cooling," provides instruction to manually restore RHR shutdown cooling and isolate containment. Throughout the event, the reactor pressure remained significantly below the shutoff head of all of the Low Pressure Coolant Injection (LPCI) [BO] and Low Pressure Core Spray (LPCS) [BM] pumps, thereby, ensuring that sufficient capability was available to maintain reactor water inventory.

The margins in time-to-boil, Technical Specification provisions, available compensatory measures, and crew preparation mitigated the potential safety consequences associated with the RHR SDC isolation.

This event is only applicable to shutdown conditions since SDC can only be used during shutdown at low reactor pressure. If this event occurred with a shorter time to boil, operators would have been able to respond more quickly to restore SDC. Nonetheless, the safety consequences would be limited with the reactor vessel closed and alternate methods of decay heat removal available.

Based on NRC Integrated Inspection Report 05000397/2005004, this event is reportable in accordance with 10 CFR 50.73 (a)(2)(v)(B).

Further Corrective Actions

Energy Northwest will revise procedures SOP-RHR-SDC-BYPASS and PPM 2.7.6 to designate disconnect RHR-42-8BA2A rather than RHR-DISC-V/9 to disable RHR-V-9.

Energy Northwest will identify and perform a technical review of all SDC-related procedures for technical accuracy.

Energy Northwest will reinforce expectations for technical accuracy and completeness with Procedure Sponsors and Qualified Procedure Reviewers and ensure that clear and specific guidance is provided to those performing procedure validations.

The extent of condition could extend to other infrequently used procedures. Energy Northwest will identify appropriate infrequently used procedures and perform a further review to identify and correct any technical inadequacies of consequence.

Similar Events

There have been two reported isolations of SDC at Columbia within the past 5 years.

LER 2003-003 reported an RHR SDC isolation due to the closure of the inboard RHR SDC primary containment isolation valve RHR-V-9. This event occurred during a planned maintenance activity and was caused by maintenance personnel performing work on the wrong relay when replacing a relay wire lug.

LER 2003-005 reported an RHR SDC isolation due to the closure of the outboard RHR SDC primary containment isolation valve RHR-V-8. This event occurred during a planned surveillance test and was caused by an inadequate surveillance procedure.

EIIS Information (Denoted as [XX])

Text Reference	System	Component
Reactor Recirculation Pump, RRC-P-1A	AD	P
Reactor Pressure Vessel	AD	RPV
High Pressure Core Spray	BG	
Low Pressure Core Spray	BM	
Low Pressure Safety Injection	BO	
RHR-DISC-V/9	BO	DISC
RHR-42-8BA2A	BO	DISC
RHR SDC Isolation Valve, RHR-V-9	BO	ISV
RHR SDC Pump, RHR-P-2B	BO	P
Reactor Protection System	JC	