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NP-33-98-002-01

Docket No. 50-346

License No. NPF-3

November 7, 2003

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Ladies and Gentlemen:

LER 1998-002-01
Davis-Besse Nuclear Power Station, Unit No. 1
Date of Occurrence – April 10, 1998

Enclosed please find Supplement 1 to Licensee Event Report (LER) 1998-002, which is being submitted to provide additional information regarding an event that resulted in manual actuation of the Reactor Protection System following demineralizer resin blockage of the letdown line. Notification of this event was provided on April 10, 1998 (Event No. 34056). This LER is being reported pursuant to 10 CFR 50.73(a)(2)(iv)(A) as an event that resulted in manual actuation of the Reactor Protection System.

This Supplement addresses corrective actions that have since been implemented. The content of the LER has been expanded to enhance the presentation of the various aspects of the event, including a description of a brief Reactor Coolant System cooling transient that did not exceed plant limits. Commitments associated with this LER are listed in the Attachment.

Very truly yours,

Lew W Myers

PSJ/s

Enclosures

IE22

cc: Regional Administrator, USNRC Region III
DB-1 Project Manager, USNRC
DB-1 NRC Senior Resident Inspector
Utility Radiological Safety Board

COMMITMENT LIST

The following list identifies those actions committed to by the Davis-Besse Nuclear Power Station in this document. Any other actions discussed in the submittal represent intended or planned actions by Davis-Besse. They are described only as information and are not regulatory commitments. Please notify the Manager - Regulatory Affairs (419-321-8450) at Davis-Besse of any questions regarding this document or associated regulatory commitments.

COMMITMENTS	DUE DATE
1. Determine the failure mechanism for Makeup and Purification Demineralizer #3.	1. Completed
2. Perform flushing of the letdown flow path, as necessary.	2. Completed
3. Provide further evaluation or testing of valve MU4.	3. Completed
4. Evaluate ICS for a trip at low power to determine if operating practices can be enhanced.	4. Completed

NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
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 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@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 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
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4. TITLE
Plant Trip Due to High Pressurizer Level as a Result of Loss of Letdown Capability

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
04	10	1998	1998	-- 002 --	01	11	7	2003	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1
10. POWER LEVEL: 31

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

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

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Peter S. Jordan – Regulatory Affairs	(419) 321-8260

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	CB	FDM	I020	YES	X	JA	LC	B015	NO
X	CB	ISV	L200	NO					

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

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

On April 10, 1998, at 2000 hours, with the plant operating at approximately 72 percent power, a scheduled plant shutdown was initiated. At 2124 hours, Makeup and Purification Demineralizer (PD) #3 was placed in service. Makeup Filter (MUF) differential pressure alarms were received in the Control Room shortly thereafter. In response, the MUF in service was swapped with the standby filter. Following additional alarms, the letdown system was isolated, and a rapid plant shutdown was initiated. Due to the letdown flowpath being isolated, Pressurizer (PZ) level increased, and at 2231 hours, the reactor was manually tripped from 31 percent power due to PZ level exceeding the maximum administrative level of 290 inches. The manual trip was required by high PZ level due to filter restrictions and loss of letdown capability. A cause analysis concluded the filter restrictions resulted from corrosion failure of PD #3 internal screens that allowed resin release into downstream piping. The screens were repaired in March 2002, and a resin control program was implemented in June 2003. Initial notification to the NRC was made at 2327 hours on April 10, 1998, in accordance with 10 CFR 50.72(b)(2)(ii). This Licensee Event Report is being submitted in accordance with 10 CFR 50.73(a)(2)(iv)(A), an event that resulted in manual actuation of the Reactor Protection System.

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

DESCRIPTION OF OCCURRENCE:

On April 10, 1998, at 2000 hours, with the plant in Mode 1 operating at approximately 72 percent rated thermal power, a scheduled plant shutdown was initiated to start the Eleventh Refueling Outage (11RFO). At 2124 hours, Makeup and Purification Demineralizer (PD) #3 [CB-FDM] was placed in service at the request of the Chemistry Unit and Reactor Coolant System (RCS) letdown flow was increased to approximately 100 GPM. At 2127 hours, the Control Room (CTRM) annunciator alarm, Letdown or Makeup Filter Differential Pressure High, was received along with the computer alarm, Reactor Coolant Makeup Filter Differential Pressure High. Subsequently, a Letdown Pressure High annunciator alarm was received in the CTRM at 2129 hours and the Letdown System Relief Valve (MU1890) [CB-RV] lifted to relieve the letdown pressure to the Reactor Coolant Drain Tank. Makeup Filter #1 [CB-FLT] was placed in service, Makeup Filter #2 was removed from service at 2127 hours, and the alarms cleared at 2130 hours. At 2131 hours, after swapping Makeup Filters, the same annunciator and computer alarms were again received for High Makeup Filter differential pressure, followed at 2133 by the Letdown Pressure High annunciator. In response to the high letdown pressure alarm, valve MU4, Letdown Block Orifice Isolation Valve [CB-ISV] was closed at 2137 hours in accordance with the alarm procedure. Valve MU4 failed to close completely on demand from the control room, and valve MU3, Letdown Isolation Valve [CB-ISV], was closed to allow relief valve MU1890 to reseat. Valve MU4, a motor operated valve, was then manually closed locally. At 2143, Pressurizer (PZ) [AB-PZR] level had increased to approximately 240 inches as a result of the letdown flowpath being isolated, and the annunciator, Pressurizer Level High, was received in the CTRM. The procedure DB-OP-02522, "Small RCS Leaks," was entered at 2145 hours due to relief valve MU1890 lifting. At 2157 hours, an attempt was made to control PZ level by opening MU6, Letdown Flow Control Valve [CB-FCV], and diverting letdown to the Clean Waste System [WD]. The letdown high pressure alarm was again received along with the Clean Waste Primary Filter Differential Pressure High alarm which indicated that the Clean Waste Primary Demineralizer Filter [WD-FLT] was restricted. Control valve MU6 was then closed to isolate letdown.

At 2200 hours, a rapid plant shutdown was initiated in accordance with procedure DB-OP-02504, "Rapid Shutdown." Technical Specification (TS) 3.1.2.2, Action a. for the boric acid addition flow path from the concentrated boric acid storage system [CA] to the RCS, was entered at 2207 hours due to the loss of the flowpath through the Makeup Filters. PZ level continued to increase while further attempts were made to control RCS inventory. At 2231 hours, the reactor was manually tripped from 31 percent reactor power because PZ level had exceeded the maximum administrative level of 290 inches. The PZ level at the time of the manual reactor trip was 304.8 inches. The maximum PZ level recorded post-trip was 306.7 inches, as indicated by computer point L768.

The Reactor Protection System (RPS) functioned properly to open the Control Rod Drive Trip Breakers. All control rods inserted on the reactor trip as designed. All four RPS channels tripped on RCS low pressure as expected after the reactor

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

DESCRIPTION OF OCCURRENCE (continued):

trip. The Steam Generator (SG) outlet pressure increased due to the closing of the Main Turbine Stop Valves. The Turbine Bypass Valves and the Atmospheric Vent Valves (AVVs) opened, and several Main Steam Safety Valves (MSSVs) lifted as designed to limit secondary system pressure. The MSSVs and the AVVs closed as SG outlet pressure decreased. The Turbine Bypass Valves controlled SG outlet pressure at the post-trip setpoint (~995 psig). The SG pressures decreased below the post-trip setpoint (SG #1 decreased to 969 psig and SG #2 decreased to 982 psig) 52 minutes after the trip due to low reactor core decay heat output and secondary system steam load demands. Main steam loads were reduced when Main Feedwater Pump #1 was removed from service at 2330 hours.

Following the reactor trip, SG #1 Startup Feedwater Control Valve (SP7B) [SJ-LCV] initially was demanded by the Integrated Control System (ICS) [JA] to close to the 27 percent open position. The valve was then released on level control, and the valve returned to a full open position in 13 seconds. SG #1 level dropped to a minimum of 24.7 inches due to the steam load on SG #1 and the initial closure of valve SP7B. The Steam and Feedwater Rupture Control System (SFRCS) [JB] low SG level trip setpoint is 23.5 inches.

At 2330 hours, the 235 lb. Auxiliary Steam (AS) header was placed onto the Auxiliary Boiler. At 2335 hours, an operator noticed a hot spot on the boiler shell. Due to concerns with boiler integrity, the Auxiliary Boiler was manually tripped at 2358 hours, and the 235 lb. AS header was aligned back to the Main Steam System being supplied from SG #1. The increased steam load from AS on #1 SG caused steam pressure and water level to decrease. The Startup Feedwater Control Valve SP7B opened in response to the demand and caused an overfill of the #1 SG. The operator took manual control of the valve and closed it at 0006 hours on April 11, 1998. Abnormal procedure DB-OP-02526, "SG Overfill," was entered. The steam pressure decrease and overfill resulted in a cooling of the RCS and DB-OP-02000, "Emergency Procedure," Section 7 - Overcooling, was entered at 0011 hours on April 11, 1998. Actions were taken to reduce steam loads to minimize cooling of the RCS. Conditions were stabilized, and at 0023 hours DB-OP-02000 was exited due to the cooling transient being terminated. At 0026 hours, DB-OP-02526 was exited for the SG overfill.

Initial notification (Event No. 34056) of the Nuclear Regulatory Commission was made via the Emergency Notification System at 2327 hours on April 10, 1998, in accordance with 10CFR50.72(b)(2)(ii), for an event that resulted in manual actuation of the Reactor Protection System. As the result of the January 2001 revision to 10CFR50.72, this notification would now be made in accordance with 10CFR50.72(b)(2)(iv)(B). This Licensee Event Report is being submitted in accordance with 10CFR50.73(a)(2)(iv)(A)as an event that resulted in manual actuation of the Reactor Protection System.

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DESCRIPTION OF OCCURRENCE (continued):

Technical Specification 3.1.2.2 was exited on April 11, 1998, at 0245 hours after Makeup Filter #1 was changed and returned to service to re-establish a concentrated boric acid addition flowpath. The boric acid addition flow path from the concentrated boric acid storage system was inoperable for approximately 4 hours and 38 minutes which was within the TS allowed action time limit of 72 hours.

APPARENT CAUSE OF OCCURRENCE:

The reason that the Reactor was manually tripped is that the PZ level exceeded the procedurally specified high level limit of 290 inches when in Modes 1 and 2. The high level condition in the PZ was due to loss of letdown capability as a result of the Makeup Filters and Primary Demineralizer Filter being restricted. The filters were restricted after PD #3 was put in service. Subsequent inspection of PD #3 revealed that its internals had degraded. This resulted in demineralizer resin being transported out of the PD #3 and into the filter housings. The screen mesh that wraps around the laterals in the bottom of the demineralizer failed due to extensive pitting corrosion and material deficiencies which allowed the resin breakthrough. A metalurgical analysis conducted by Exelon Power Labs indicated that sulfur compounds, which caused a low pH, were the likely cause of the pitting. The likely source of the sulfur compounds was attributed to the degradation of the cation resin beads due to the resin's partially spent condition and extended radiation exposure to activated corrosion products.

Prior to the reactor trip, operators attempted to close MU4 from the control room. MU4 failed to fully close on this remote demand. Subsequent troubleshooting of the valve did not identify any abnormal characteristics. Therefore, the cause of the failure to close was indeterminate. During preventive maintenance tests conducted since, the valve stroked normally.

The apparent sluggish response by SP7B was evaluated. The valve's response immediately following the reactor trip was attributed to a drop in differential pressure (dP) across valve SP7B which reduced the ability of the valve to control SG level. The sluggish response of SP7B following the trip of the Auxiliary Boiler was attributed to the realignment of steam loads and not due to a tuning or hardware problem. When the steam loads were transferred, the dP across the valve was rapidly reduced. The valve went full open until the valve dP was restored. As the steam loads decreased, the valve dP recovered, but SG #1 level had overshot the setpoint.

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ANALYSIS OF OCCURRENCE:

This event had no significance to the health and safety of the public. There were no safety concerns identified as a result of this manual reactor trip. The Reactor Protection System and Control Rod Trip Breakers functioned properly and all control rods inserted on the reactor trip as designed. The Turbine Bypass Valves, AVVs, and MSSVs functioned as designed. There were no systems or components inoperable at the initiation of the plant shutdown for the 11RFO that contributed to the event.

The failure of PD #3 had minimal safety significance. The plant was successfully shut down after the manual reactor trip, and the Letdown System was returned to normal operation. The failure of non-safety grade valve MU4 to initially close completely was of minimal consequence. Valve MU3, located upstream of MU4 was closed to isolate the letdown system as needed.

The maximum post trip PZ level observed was 306.7 inches. The 305 inch PZ level upper limit of Technical Specification (TS) 3.4.4 was exceeded for approximately 8 seconds. However, since the plant had tripped, the TS 3.4.4 upper limit for PZ level was no longer applicable. Post trip shrinkage of RCS inventory quickly lowered PZ level below 305 inches.

Startup Feedwater Control Valve SP7B for SG #1 responded to the ICS demand signal properly immediately following the trip. The plant trip occurred while the ICS was controlling SG levels at the low level limit setpoint of 40 inches. The ICS Rapid Feedwater Reduction (RFR) circuit initiated as expected. However, valve SP7B started to close towards its RFR target setpoint of 18 percent, which should not have occurred because SG #1 was already on ICS level control. This response was attributed to low dP across SP7B which reduced the ability of the valve to control SG level. Control valve SP7B was then immediately released, returned to level control and stroked to its full open position in approximately 13 seconds. Main Feedwater Pump (MFP) #1 was in service at the time and was being supplied steam from SG #1. The effect of tripping from this low power, with MFP #1 in service, and valve SP7B initially closing to 27 percent, was a lowering of SG #1 level below the normal low level limit control setpoint. As the MFP increased speed and SG pressure decreased with valve SP7B open, the proper post trip SG level of 40 inches was achieved within 30 seconds. Although the SG #1 level decreased below the normal post trip setpoint, SG level was recovered, and the SFRCS was not challenged. SG pressure decreasing below setpoint, 52 minutes after the reactor trip, was due to the plant trip occurring from low power with a decreased amount of decay heat available. There was no overheating or overcooling of the RCS during this portion of the post trip sequence.

Approximately 90 minutes following the manual trip of the reactor, while still within TS 3.1.2.2.a action statement, the AS header was realigned from the Auxiliary Boiler to SG #1. As a result of the increased steam demand created

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ANALYSIS OF OCCURRENCE (continued):

when the operators realigned the AS header to the Main Steam System, SG #1 level decreased below the 40-inch level setpoint. SP7B responded to the low level by going 100% open, thereby over-feeding SG #1. The over-feed was terminated by the operator taking SP7B to manual control and closing the valve until SG #1 level was returned to the desired 40-inch level. The SG #1 over-feed contributed to a cooling of the RCS. A review of the cooling data for the RCS, PZ, and SG determined that there were no plant cooldown rate limits exceeded. The maximum RCS cooldown rate was 1.5 degrees F per minute for five minutes which is less than the TS 3.4.9.1.b limit of 100 degrees F per hour (1.67 degrees F/minute).

CORRECTIVE ACTIONS:

PD #3, manufactured by Illinois Water Treatment Company, was isolated following the trip and tagged out of service. Makeup Filter #1 was replaced and Makeup Filter #2 was blown down with nitrogen to the Spent Resin Storage Tank. Makeup Filter #2 was later changed. The letdown flowpath was returned to service and the plant shutdown for the 11RFO was continued.

An evaluation of the letdown flowpath of the Makeup and Purification System to identify portions of the system that may require flushing to support plant startup following the 11RFO was performed. Additional flushing was completed on May 18, 1998, prior to entry into Mode 2 for Cycle 12.

Prior to entry into Mode 2 for Cycle 12, troubleshooting of valve MU4, which does not have a safety function, was performed under static system conditions to determine why it failed to fully close. Due to the valve's configuration, normal flowpath being through the valve from above the plug, static testing would not include additional forces that would normally assist valve closure. Additional stroking and monitoring of MU4 was performed during Mode 3 at full system pressure. No abnormal valve characteristics were observed during this test or during subsequent tests performed since the event. Therefore, the cause of the valve's failure to fully close is indeterminate.

The Rapid Feedwater Reduction relays that cause Startup Feedwater Valve SP7B to go to its target position immediately following the reactor trip were bench tested during the 11RFO. The relays performed their function. An evaluation of the apparent ICS post-trip sluggish response of SP7B for a trip from low power was performed to determine if any ICS tuning or operating practices could be enhanced. It was determined that the apparent sluggish response of SP7B was due to the plant not being in an operating mode with valve dP being controlled at a setpoint. The startup data for Cycle 12 was reviewed, and the SG level control was normal. No tuning or hardware deficiencies were identified. During 13RFO, as routine preventive maintenance, an entire rebuild of the valve actuator was

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CORRECTIVE ACTION (continued):

performed, and accessories that support operation of SP7B were either replaced or rebuilt. Diagnostic testing was performed to confirm the acceptability of the valve internals and its packing.

In order to assure other demineralizers were not exposed to the same degrading low pH environment as PD #3, stagnant water samples were obtained and analyzed from Primary Demineralizers #1 and #2, the Clean Waste Polishing Demineralizer, and the Waste Polishing Demineralizer. The pH values and sulfate concentrations were found to be within acceptable ranges.

An evaluation of Cycle 12 operation without the cation Makeup and Purification Demineralizer #3 was completed and documented in Potential Condition Adverse to Quality Report 98-0553. It was determined that operation during Cycle 12 with only mixed bed Makeup and Purification Demineralizers #1 and #2 available was acceptable. Subsequently, due to significant radiation dose considerations, the repair activity was extended to 13RFO. The action to maintain PD #3 out of service for Cycle 13 was evaluated by Safety Evaluation (SE) 98-0037, Revision 1, to be acceptable.

In March 2002, during 13RFO, the retention elements of PD #3 were replaced with an upgraded design and by upgrading the lateral mesh material from 304 to 316 stainless steel. This offers enhanced resistance to pitting.

To prevent recurrence of this condition, a resin control program has been instituted for the operation of the DBNPS primary system demineralizers. DB-CH-04052, "Maintenance of Primary System Demineralizers," became effective June 16, 2003. This procedure provides a tracking mechanism to ensure that the demineralizer resins are changed out prior to any degradation.

FAILURE DATA:

There have been no previous plant trips at the DBNPS as a result of high PZ level.

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

NP-33-98-002-01	PCAQR 98-0529	PCAQR 98-0531
	PCAQR 98-0533	PCAQR 98-0534
	PCAQR 98-0553	CR 99-0869
	CR 02-04296	CR 03-00357