FENOC

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February 16, 2010
L-10-023

10 CFR 50.73

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

SUBJECT:
Davis-Besse Nuclear Power Station
Docket Number 50-346, License Number NPF-3
Licensee Event Report 2009-002

Enclosed is Licensee Event Report (LER) 2009-002, "Vibration Induced Failure of Temperature Instrument Results in Operation above Licensed Power Limit." This LER is being reported in accordance with 10 CFR 50.73(a)(2)(i)(B) as an operation or condition prohibited by the Technical Specifications.

There are no regulatory commitments contained in this letter or its enclosure. The actions described represent intended or planned actions and are described for information only. If there are any questions or if additional information is required, please contact Mr. Dale R. Wuokko, Manager, Site Regulatory Compliance, at (419) 321-7120.

Sincerely,

Clark A. Price for

Barry S. Allen

Enclosure: LER 2009-002-00

cc: NRC Region III Administrator
NRC Resident Inspector
NRR Project Manager
Utility Radiological Safety Board

IE22
NRR

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 8/31/2010

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Davis-Besse Nuclear Power Station	05000346	1 OF 5

4. TITLE
Vibration Induced Failure of Temperature Instrument Results in Operation above Licensed Power Limit

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
12	16	2009	2009	- 002 -	00	02	16	2010	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

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Gerald M. Wolf, Supervisor, Nuclear Compliance	(419) 321-8001

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
B	IO	TE	R369	N					

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

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

On December 16, 2009, with the plant in Mode 1 at approximately 100 percent power, it was identified that one of the inputs to the secondary heat balance for steam temperature was indicating approximately 8 degrees F lower than the input for the opposite steam line temperature indication. Based on heat balance calculations, this 8 degree F difference resulted in an indicated power of approximately 0.45 percent full power less than actual reactor thermal power. This 8 degree F difference resulted from a failed resistance temperature detector (RTD). The RTD failed due to the vibration of the main steam lines because the RTD was not long enough to ensure proper compression of the sensing element in the thermowell. A new RTD was installed, and this RTD will be replaced every two years until an RTD of appropriate length with better resistance to vibration is installed.

A review of historical plant computer data showed this temperature difference existed since April 2006, and resulted in the plant operating at a maximum 8 hour average reactor power of 100.374 percent, which exceeded the maximum analyzed power level allowed by the Technical Specifications by 0.004 percent on one occasion. This issue is being reported in accordance with 10 CFR 50.73(a)(2)(i)(B) as an operation prohibited by the Technical Specifications.

NARRATIVE

Energy Industry Identification System (EIIS) codes are identified in the text as [XX]. All numerical values given in the discussion below are approximate, based on accuracy of the affected instruments and the averaging of historical computer data.

DESCRIPTION OF EVENT:

Applicable Technical Specifications:

Technical Specification (TS) Limiting Condition for Operation (LCO) 3.2.1 for Regulating Rod Insertion Limits requires that the regulating groups be within the physical insertion, sequence, and overlap limits specified in the Core Operating Limits Report (COLR) while in Modes 1 and 2. With regulating rod groups inserted in the restricted operation region, Surveillance Requirement 3.2.5.1 is required to be performed once per 2 hours if reactor thermal power is greater than 20 percent, and the regulating rod groups are required to be restored to within limits in 24 hours from discovery. The corresponding COLR Figures 2a and 2b for operation with 4 Reactor Coolant Pumps show a maximum 100.37 percent rated thermal power (including instrument error) as the top of the acceptable operation region.

TS LCO 3.2.3 for Axial Power Imbalance Operating Limits requires that the Axial Power Imbalance be within the limits specified in the COLR while in Mode 1 greater than 40 percent rated thermal power. With the imbalance not within the specified limits, Surveillance Requirement 3.2.5.1 is required to be performed once per 2 hours, and the Axial Power Imbalance is required to be reduced to within limits in 24 hours. The corresponding COLR Figures 4a, 4b, and 4c for operation with 4 Reactor Coolant Pumps show a maximum 100.37 percent rated thermal power (including instrument error) as the top of the permissible operating region.

Initial Plant Conditions:

On December 16, 2009, the Davis-Besse Nuclear Power Station (DBNPS) was operating in Mode 1 at approximately 100 percent power.

Event Description:

On December 16, 2009, the System Engineer for the Non-Nuclear Instrument System [IO] identified that one of the inputs to the secondary heat balance for the steam temperature at the inlet to the high pressure turbine was indicating 8 degrees Fahrenheit (F) lower than the input for the opposite steam line temperature indication. The value for plant computer point T476 for Steam Generator 2 indicated 583.7 degrees F compared to 591.8 for computer point T477 for Steam Generator 1. Alternate indications for the steam temperature at the outlet of the Steam Generators were consistent with the indications from computer point T477. Based on heat balance calculations, this 8 degree F difference resulted in a calculated/ indicated heat balance power of 0.45 percent reactor thermal power less than actual reactor thermal power.

A review of historical data from the plant computer showed that this difference has existed since the beginning of the previous operating cycle (April 2006). After replacement of both temperature elements providing steam temperature to the secondary heat balance, the temperature error was determined to be 5.5 to 6.0 degrees F instead of the initially-identified 8 degrees F. From the historical data review, which included actual operating conditions, the maximum one hour average reactor power was 100.475 percent, and the maximum 8 hour average reactor power was 100.374 percent.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Davis-Besse Unit Number 1	05000346	2009	-- 002 --	00	3 OF 5

NARRATIVE

CAUSE OF EVENT:

In 2002, the two temperature elements (TESP15A and TESP15B) [IO-TE] providing indication of the steam temperature at the inlet to the high pressure turbine were replaced with new Rosemount series 0078 Platinum Resistance Temperature Detectors (RTDs) along with matched transmitters to provide a greater accuracy than the previously installed RTDs. This replacement was done as part of the modification to install a Leading Edge Flow Meter and improve the accuracy of the plant power calorimetric measurement to allow an increase of 1.63 percent rated thermal power. The changes to the Operating License and Technical Specifications for this 1.63 percent rated thermal power increase were approved by the NRC on June 30, 2008 and plant power was increased accordingly on July 14, 2008.

The Rosemount series 0078 RTD is a wire wound device with a resistance that changes as a function of temperature. The wire coil is encapsulated in a spring loaded sensor assembly, and the sensor is installed in a thermowell with the tip compressed against the end of the thermowell via a spring to provide an accurate measurement of the temperature along with a faster response time and better vibration resistance.

The RTD providing the erroneous indication (TESP15A) was removed from the system and sent for failure analysis, which determined that the temperature element had been damaged by vibration, resulting in the output shifting low. The Main Steam piping where the temperature element is installed is subject to constant vibration while the plant is in operation. The damage was caused by the piping vibration in conjunction with the incorrect length of the RTD. The RTD is approximately the same length as the thermowell in which it is installed. However, the RTD should be approximately 0.5 inches longer than the thermowell to ensure proper compression of the sensing element. Without the proper compression, additional movement of the RTD in the thermowell is likely, contributing to the vibration-induced failure. The RTDs installed in 2002 were the same length as the originally installed RTDs, so the short RTD length appears to be a latent error from original plant construction. The original Rosemount 104AFP RTDs (installed prior to 2002) were less accurate than the new RTDs but appear to have been much more rugged, so the length of the thermowell and vibration did not adversely affect their operation.

ANALYSIS OF EVENT:

As discussed previously, actual plant operating conditions were reviewed from the plant computer and the heat balance power was recalculated based on the temperature input error to determine maximum power levels. The maximum one hour average reactor power was 100.475 percent and the maximum 8 hour average reactor power was 100.374 percent, both occurring on August 16, 2008. Comparing the recalculated heat balance power to the COLR figures for Rod Insertion Limits and Axial Power Imbalance indicates the reactor was operated greater than the maximum power of the acceptable operation region for a total of 333 hours over the total 3769 hour period from July 17, 2008 to December 21, 2008, based on the one hour average reactor power calculation. Based on the 8 hour average reactor power calculation, the reactor was operated greater than the maximum power of the acceptable region for one (1) 8 hour period on August 16, 2008.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Davis-Besse Unit Number 1	05000346	2009	-- 002 --	00	4 OF 5

NARRATIVE

ANALYSIS OF EVENT: (continued)

Eight (8) hour periods were analyzed since these are equivalent to a shift. Under NRC Regulatory Issue Summary 2007-21, Revision 1, "Adherence to Licensed Power Limits," (dated February 9, 2009) and the referenced Nuclear Energy Institute (NEI) guidance, core thermal power average for a shift is not to exceed the licensed power level.

Prior to use of the Leading Edge Flow Meter, the maximum authorized steady-state reactor core power levels, per DBNPS Operating License Condition 2.C.(1) was 2772 Megawatts thermal (MWth). The maximum analyzed steady-state reactor core power levels, including uncertainties, was 102 percent of 2772 MWth, or 2828 MWth. With the change to the Operating License to allow use of the Leading Edge Flow Meter, while the maximum authorized steady-state reactor core power level was increased to 2817 megawatts thermal, the maximum analyzed steady state reactor core power level remained the same at 2828 MWth. For the time period of April 2006 to December 2009 when the temperature error existed, the maximum eight hour average reactor power exceeded the analyzed power level for only one eight hour time period and then by only 0.004 percent of rated thermal power; therefore, this issue had very low safety significance.

Reportability Discussion:

TS LCO 3.2.1 for Regulating Rod Insertion Limits requires that the regulating groups shall be within the physical insertion, sequence, and overlap limits specified in the COLR while in Modes 1 and 2. The corresponding COLR Figures 2a and 2b for operation with 4 Reactor Coolant Pumps show a maximum 100.37 percent rated thermal power (including instrument error) as the top of the acceptable operation region. TS LCO 3.2.3 for Axial Power Imbalance Operating Limits requires that the Axial Power Imbalance be within the limits specified in the COLR while in Modes 1 greater than 40 percent rated thermal power. The corresponding COLR Figures 4a, 4b, and 4c for operation with 4 Reactor Coolant Pumps show a maximum 100.37 percent rated thermal power (including instrument error) as the top of the permissible operating region. Because the plant operated at a maximum of 100.374 percent rated thermal power for an 8 hour average, this was in excess of the 100.37 percent rated thermal power limit specified in the COLR, and represents a condition prohibited by TS LCO 3.2.1 and 3.2.3. Therefore, this issue is reportable per 10 CFR 50.73(a)(2)(i)(B) as operation or condition prohibited by the Technical Specifications.

CORRECTIVE ACTIONS:

Following discovery of this condition, reactor power was lowered to less than 99.5 percent on December 17, 2009, to account for the faulty RTD indication. A conservative value was manually input to the heat balance calculation to allow for the faulty RTD to be removed from the system for troubleshooting, and reactor power restored to 100 percent. The faulty RTD was replaced and the temperature transmitter calibrated on December 19, 2009. The other RTD (TESP15B) on the opposite steam side monitoring steam temperature at the inlet to the high pressure turbine was replaced on January 18, 2010.

These two RTDs will be replaced with RTDs of the appropriate length for the installed thermowells as well as with a type of RTD that is more resistant to vibration. As an interim measure, preventive maintenance activities will be revised to replace the 2 RTDs every 2 years until they are replaced with a more robust type.

LICENSEE EVENT REPORT (LER)
CONTINUATION SHEET

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Davis-Besse Unit Number 1	05000346	2009	-- 002 --	00	5 OF 5

NARRATIVE

PREVIOUS SIMILAR EVENTS

In 2004 a failure of TESP15B was identified, but no evaluation of the failure was performed. The RTD had been installed for 2 years at the time of the failure. The procedures in place at that time did not require a failure evaluation be performed for equipment designated as non-critical, such as these RTDs. Current procedures require that the failure of critical or non-critical components be evaluated and an equipment apparent cause evaluation and failure mode analysis be performed.