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Gentlemen:

DOCKET 50-301
LICENSEE EVENT REPORT 94-001-01
POTENTIAL FEEDWATER FLOW MEASUREMENT INACCURACIES
POINT BEACH NUCLEAR PLANT UNIT 2

Enclosed is a supplement to Licensee Event Report 94-001-00 for Point Beach Nuclear Plant Unit 2. This report is being submitted to provide additional information about corrective actions for this event. The supplemental information is provided after the original text of this Licensee Event Report.

Please contact us if there are any questions.

Sincerely,

for Bob Link
Vice President
Nuclear Power

KVA/jg

cc: NRC Resident Inspector
NRC Regional Administrator

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
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ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Point Beach Nuclear Plant, Unit 2	05000301	1 OF 8

TITLE (4)
Potential Feedwater Flow Measurement Inaccuracies

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
03	23	94	94	-- 001 --	01	07	29	94	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)					
POWER LEVEL (10)	98.5	20.402(b)		20.405(c)		50.73(a)(2)(iv)	73.71(b)
		20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)	OTHER
		20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)	(Specify in Abstract below and in Text, NRC Form 366A)
		20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)	
		20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Ken Arneson, Senior Engineer, Licensing	(414) 221-3362

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

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

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

On March 23, 1994, with both units operating at full power, we identified conditions which indicate that feedwater flow in Unit 2 may have been underestimated by approximately 2% since the beginning of the current operating cycle. Attempts to verify this indication have not been completed due to equipment problems. The cause of this potential underestimation of feedwater flow is believed to be due to a degradation in the signals from the Leading Edge Flow Meter transducers. As a conservative measure, an updated Reactor Thermal Output calorimetric calculation was performed based on the potential 2% underestimation of feedwater flow. The results of this calculation indicate that reactor power may have been 2% higher than indicated and one channel of Overtemperature Delta T and Overpower Delta T protection may have been inoperable. Power range nuclear instrument gains, and Overtemperature Delta T and Overpower Delta T setpoints were adjusted accordingly. We will continue our evaluation of the potential underestimation of feedwater flow and report the results in a supplemental Licensee Event Report.

Event Description:

On March 23, 1994, with both units operating at full power, we identified conditions which indicate that feedwater flow in Unit 2 may have been underestimated by approximately 2% since the beginning of the current operating cycle. The current operating cycle for Unit 2, Cycle 20, began on October 30, 1993.

A team of engineers was established in early 1993 to evaluate an apparent difference in the thermal efficiencies between Point Beach Nuclear Plant (PBNP) Unit 1 and 2. As part of this evaluation, a vendor was contracted to conduct an assessment of the PBNP feedwater flow measurement system. The assessment centered around the accuracy of the Leading Edge Flow Meter (LEFM).

PBNP currently uses a Model 601 LEFM. The contractor temporarily installed a Model 8300 LEFM electronics unit on December 15-17, 1993 to provide an independent measure of the feedwater flow in both units. The vendor presented the results of their assessment to plant management on March 15, 1994. Their assessment indicated that the Model 601 LEFM is accurately measuring Unit 1 feedwater flow, but was underestimating Unit 2 feedwater flow by approximately 1.0%.

After the vendor completed the data collection, but while the Model 8300 LEFM was still installed, we discovered a small amount of feedwater flow was bypassing the Unit 2 LEFM but not the venturis. The flow was occurring through the Unit 2 'A' train main feed regulation valve (MFRV) bypass line, which taps off the main feedwater header upstream of the LEFM. The air-operated valve in the bypass line, 2CS-480, being subject to flow induced vibration, had been cracked open by operators to mitigate the vibration. Once discovered, flow through the bypass line was isolated by shutting manual isolation valves in the line. By comparing the Model 8300 LEFM flow measurement before and after isolating the bypass line, we measured the bypass flow at 0.25% of total feedwater flow. This bypass flow therefore increases the underestimation of feedwater flow, as measured by the Model 8300 LEFM, from 1.0% to 1.25%.

On March 22 and 23, 1994, the Model 601 LEFM was reconnected. A Feedwater Flow Correction Factor (FFCF) test was performed the morning of March 23, 1994, in accordance with Reactor Engineering Surveillance Procedure (RESP) 1.4, "PPCS Feedwater Flow Correction," to verify proper operation of the Model 601 LEFM and to determine if the Unit 2 FFCF needed to be updated. Valve 2CS-480 was verified to have been shut during the conduct of this test. The results of the test indicated that

the Model 601 LEFM may have been underestimating actual feedwater flow in Unit 2 by approximately 1.84%. Attempts to verify this indication have not been completed as the Model 601 LEFM has been spiking since the March 23, 1994, test and is producing variable output.

As the Model 8300 LEFM has more advanced signal processing capabilities than the Model 601 LEFM, we believe the underestimation of feedwater flow by 1.25% is more accurate. However, the Model 8300 LEFM was used only on a temporary basis to gage the accuracy of the Model 601 LEFM. It is not a permanently installed system and cannot be used to correct the venturi flow measurement for use in the calorimetric calculation. Therefore, until we can validate the Model 8300 LEFM data, we are conservatively assuming the 1.84% underestimation of feedwater flow. As a result, based on the Unit 2 FFCF determined at the beginning of the Cycle 20, feedwater flow could have been approximately 2% more than indicated since that time.

Reactor Thermal Output (RTO), Overtemperature Delta T (OTΔT) and Overpower Delta T (OPΔT) setpoint calculations were performed based on the assumed 2% underestimation of feedwater flow. The RTO calorimetric calculation indicated that reactor power may have been approximately 2% higher than previously indicated. With this information, the OTΔT and OPΔT setpoint calculations indicated that the current red channel OTΔT and OPΔT setpoints may have been greater than the limits given in TS 15.2.3.1(B)(4) and (5), and would therefore be inoperable.

As a precautionary measure, reactor power was reduced from approximately 98.5% to 95.5% while these calculations were being performed. Reactor power was maintained at this level until adjustments to the power range nuclear instrument gains and OTΔT/OPΔT setpoints were made. Reactor power was then returned to 99% and is being maintained at that level in order to provide sufficient margin to the new OTΔT/OPΔT setpoints.

Component and System Description:

The Model 601 LEFM installed in PBNP Units 1 and 2 (1/2FE-3110) is a digital ultrasonic flow measurement system. The LEFMs were installed during the respective unit refueling outages in 1982.

The LEFM consists of a flow measurement spool piece section in the main feedwater header for each unit and a common electronics unit with a mass flow rate display for each unit. Each spool piece contains four pairs of high temperature transducers. The feedwater flow measurement is based on the difference in transit times of upstream and downstream ultrasonic pulses between the transducers. The LEFM is used to determine

a correction factor which is applied to the feedwater venturi reading to account for corrosion product fouling of the venturi. The FFCF is normally calculated at the beginning of each operating cycle and verified approximately once a month to ensure no significant changes occur during the cycle.

The "corrected" venturi flow is used in the RTO calorimetric calculation to determine reactor power. Reactor Thermal Output is the indication by which the reactor is operated and is used to calibrate the power range nuclear instruments, calculate reactor coolant system (RCS) flow, and as the basis for the OTΔT and OPΔT setpoint calculations.

Cause:

We believe the cause of the possible underestimation of feedwater flow in Unit 2 is due to a gradual degradation in the signals from the Model 601 LEFM transducers. The Model 601 LEFM is not capable of detecting or compensating for degraded transducer signals. It was only the improved diagnostic and signal processing features of the Model 8300 LEFM which revealed several transducers with weak and/or distorted signals or no signal at all. The most likely cause of the weak signals is deformations on the surface of the transducer housings or relaxing of the transducer push rods.

Based on the trend of FFCFs for both units, it is possible the degradation began in the fall of 1991 and gradually increased to this point. Our continuing evaluation into this event will address the history of the LEFM signal degradation.

If Valve 2CS-480 was cracked open during the FFCF test performed at the beginning of Cycle 20, 0.25% of the approximately 2% difference would be due to this bypass flow. As mentioned previously, we discovered the bypass line for the Unit 2 "A" train MFRV taps off the main feedwater header upstream of the LEFM. The downstream end of the bypass line ties in to the "A" train feedwater line between the MFRV and the venturi. The drawing used for the modification to install the LEFM for Unit 2 did not accurately reflect the location of the junction of the main feedwater piping and the "A" train MFRV bypass valve piping. As such, the LEFM was installed downstream of this junction. The discrepancy was discovered when an engineer performed a walkdown of the system. Reactor engineering personnel were unaware of the Unit 2 "A" train MFRV bypass line configuration and the practice of cracking open 2CS-480.

Corrective Action:

Immediate Corrective Actions:

- Several transducers in each unit were replaced in an attempt to correct the problem with degraded signals. Useable transducer signals were achieved; however, signal to noise ratios were unsatisfactory. Several more transducer elements and push rod assemblies were then replaced and tne electrical alignment of the signals was adjusted.

- Reactor power was reduced from approximately 98.5% to 95.5% at 1423 hours on March 23, 1994, as a precautionary measure to avoid any potential overpower conditions.

- Based on the assumed 2% underestimation of feedwater flow since the beginning of Cycle 20, RTO calorimetric, OTΔT, and OPΔT setpoint calculations were performed early in the afternoon of March 23, 1994.

- The RTO calorimetric calculation indicated that reactor power may have been approximately 2% higher than previously indicated. At 1538 hours on March 23, 1994, gains on the power range nuclear instruments were adjusted accordingly.

- The OTΔT and OPΔT setpoint calculations indicated that the then existing red channel OTΔT and OPΔT setpoints may have been greater than the limits given in TS 15.2.3.1(B)(4) and (5), and were therefore inoperable. The red channel OTΔT and OPΔT signals were placed in trip within three hours while performing Instrumentation and Control Procedure (ICP) 10.11, "Delta T Setpoint Calibration at Power." ICP 10.11 was performed to input the new OTΔT and OPΔT setpoints for all channels.

- Unit 1 feedwater flow and reactor power were verified to be accurate and do not require adjustment.

- Flux mapping was performed for Unit 2 on March 24, 1994. No problems were indicated with hot channel factors.

- Through walkdowns of the main feed system, we verified that the MFRV bypass lines for both trains in Unit 1 and the "B" train in Unit 2 do not bypass the LEFM.

Short-Term Corrective Actions:

- A temporary procedure change was made to RESP 1.4 and 6.2, "Precision RCS Flow Rate Measurement," to ensure that the Unit 2 "A" train MFRV bypass line is isolated prior to conducting FFCF tests. A permanent change to this effect will be made by May 31, 1994, for both procedures.

- A Drawing Change Notice (DCN) will be submitted by April 30, 1994, to change the affected P&Ids to properly reflect the configuration of the Unit 2 "A" train MFRV bypass line.

Long-Term Corrective Actions:

- We will continue our evaluation to determine whether or not feedwater flow was actually underestimated and, if so, to what extent. If it is determined that feedwater flow was underestimated, we will evaluate the effects on RTO, OTΔT/OPΔT setpoints, core burnup calculations, and accident analyses over the associated time period. We expect to complete this evaluation by June 30, 1994. The results of this evaluation will be documented in a supplemental Licensee Event Report.

- Because the Model 601 LEFM is producing variable output, we will evaluate a long term solution for the LEFMs, including an investigation of alternate methods of determining reactor power. We expect to complete this evaluation by December 31, 1994.

Reportability:

This event is being reported in accordance with the requirements of 10 CFR 50.73(a)(2)(i), "Any operation or condition prohibited by the plant's Technical Specifications," because of the potential 2% overpower condition and inoperability of one channel of OTΔT and OPΔT protection due to the underestimation of feedwater flow since the beginning of Unit 2, Cycle 20.

Safety Assessment:

The FFCF test performed on March 23, 1994, using the Model 601 LEFM, indicated that actual feedwater flow in Unit 2 may have been approximately 2% higher than measured since the beginning of the current operating cycle. This value could not be verified because the Model 601 LEFM was spiking and producing variable output. Therefore, as a conservative measure, an updated RTO calorimetric calculation was

performed based on the potential 2% underestimation of feedwater flow. Power range nuclear instrument gains, and OTΔT/OPΔT setpoints were adjusted accordingly.

The potential 2% underestimation of feedwater flow indicated that the setpoints of one channel of OTΔT and OPΔT protection were above their TS required limits and therefore inoperable. The logic for the OTΔT/OPΔT protection is two of four. Assuming a single failure of one of the remaining three channels, two channels would have been available to provide a reactor trip signal if required.

Since the fall of 1991, Unit 2 has been operating at less than 100% rated power because of the number of plugged U-tubes in the steam generators. The average reactor power for Cycle 20 has been 98.5%. Therefore, based on the potential 2% underestimation of feedwater flow and the maximum assumed errors in the calorimetric calculation (+2%), maximum actual reactor power during this cycle could have been as high as approximately 102.5% (98.5% + 2% + 2%).

We believe the potential underestimation of feedwater flow of approximately 2% is unlikely based on data from the Model 8300 LEFM. We will continue to evaluate whether or not feedwater flow was actually underestimated and, if so, to what extent. If it is determined that feedwater flow was underestimated, we will evaluate the effects on RTO, OTΔT/OPΔT setpoints, core burnup calculations, and accident analyses over the associated time period. As flux mapping and physics testing produced normal and expected results, we do not believe the potential underestimation of feedwater flow caused any significant adverse effects on the operation or safety of the reactor core. We believe 2% is the largest amount by which feedwater flow was underestimated and, therefore, bounds any potential historical inaccuracies in the RTO calculation and OTΔT/OPΔT setpoints due to degraded LEFM transducer signals.

Similar Occurrences:

There are no known occurrences similar to this event at PBNP.

Supplemental Information:

Short-Term Corrective Actions:

- Permanent procedure changes were made to Reactor Engineering Surveillance Procedures 1.4 and 6.2 to ensure that the Unit 2 "A" train Main Feed Regulation Valve (MFRV) bypass line is isolated prior to conducting the Feedwater Flow Correction Factor (FFCF) tests. The procedure changes were effective on May 16, 1994.

- Drawing Change Notice 94-0894 was submitted on April 18, 1994, to correct the affected P&IDs to properly reflect the configuration of the Unit 2 "A" train MFRV bypass line.

Long-Term Corrective Actions:

- We have completed our evaluation of the potential underestimation of feedwater flow. The contractor who supplied the Model 8300 Leading Edge Flow Meter (LEFM) and conducted the initial evaluation of feedwater flow performed an extensive accuracy analysis of the Model 8300 LEFM test results. Calculations supporting the accuracy analysis were performed in accordance with 10 CFR 50, Appendix B. The results of the accuracy analysis validated the FFCF determined by the Model 8300 LEFM during the initial evaluation of feedwater flow. Therefore, the 1.25% feedwater flow error measured by the Model 8300 LEFM (1.0% due to signal degradation plus the 0.25% bypass flow error) is the error that applies to the calorimetric calculation. Since the magnitude of the feedwater flow measurement error was within the error assumed in the Final Safety Analysis Report (FSAR), no further evaluations were required.

- We are continuing our evaluation of the long term solution for the LEFMs. We expect to complete this evaluation by December 31, 1994.

Safety Assessment:

The magnitude of the feedwater flow error is within the error assumed in the Point Beach Nuclear Plant FSAR for the calorimetric calculation (1.25%). Hence, there was no adverse effect on plant safety. Furthermore, we believe 1.25% is the largest possible error in feedwater flow measurement and therefore bounds any potential historical inaccuracies in the Reactor Thermal Output calculation and Overtemperature Delta T and Overpower Delta T setpoints due to degraded LEFM transducer signals. Therefore, the health and safety of plant personnel and the public were not impacted by this event.