E-mol INFO

From: Nirodh Shah R3 R3 R3
To: Daniel Kimble; Frances Ramirez
Date: Thu, Feb 1, 2007 10:20 AM
Subject: Jim C. is interested in LaSalle flowrate and Bwd/BY AMAG bkgrnd info

Caldwell

Dan, Jim C. would like us to discuss our understanding of what LaSalle is proposing to do and how confident we are in their analysis. He would like this discussion sometime next week. Please mail us a copy of the 50.59 and other documentation so that Stu can take a look at it next week.

I pulled together some information re: the ultrasonic flow issue at Bwd and Byron and some info I found on the web re: Caldon flowmeters. I have attached them to this document.

In late 2002, there were two overpower issues identified at Braidwood. The first involved a non-conservative high net heat input value that had been incorporated into the calorimetric calculations. While not specifically a flowmeter issue, the value was input into the overall logic that calculates the reactor power. It also occurred concurrent with the issue discussed below. This was the subject of several Westinghouse Nuclear Safety Advisory Letters and was discussed in LER 50-457/2003-003-00.

The second involved feedwater flow pressure pulses occurring at frequencies that affected the ultrasonic flow measurement signals. This resulted in a non-conservative bias such that when reactor power was adjusted to match the calorimetric, an overpower condition resulted. This is discussed in more detail in LER 50-457/2003-002-00.

Ft Calhoun also had an issue with the AMAG system. Briefly, they had to remove the system from service (due to the Bwd and Byron events) and obtain a licensee amendment in order to complete their power uprate. However, there was no indication that Ft Calhoun had experienced problems with their AMAG.

The specific details of the Braidwood report are in the attached WP file. This input also contains the references to the BY report for the same issues. It also lists the Bwd CRs that you may be able to obtain from the licensee's computer. The attached PDF files contain some info on operating experience for power uprates (there is a section on AMAG) and some info I found on the web re: the Caldon system. A quick read of the Caldon system did not identify any real useful info other than a list of plants that have been using it.

Christian is looking at our allegation files to find out who in NRR may have reviewed this issue. Rick Skokowski, who was the Byron SRI at the time, believes that the issue was discussed with Warren Lyon.

Hope this helps....N

E-mol INFO **CC:** Bruce Burgess; Christian Scott; Jeremy Tapp; Stuart Sheldon

B-36

ATTACHMENT 1

TITLE

Operating Experience Related to Power Uprates: Abnormalities in Ultrasonic Flow Meter Instrumentation Readings

1. On August 28, 2003, Exelon informed the staff that it was reducing the operating power of Byron Units 1 and 2 by 32 MWe and 22 MWe, respectively. The decision was made following analysis of feedwater flow data derived from the Advanced Measurement and Analysis Group (AMAG) ultrasonic flow meters (UFMs) in use at Byron and Braidwood. Exelon reported that there were signal abnormalities from some of the UFMs, and on Byron 1, there were statistical differences between the total feedwater flow and the sum of the flows from the four individual feedwater lines. On September 1, 2003, the power at Braidwood Unit 2 was reduced for similar reasons. Westinghouse issued Technical Bulletin (TB) 03-6 on September 5, 2003, to inform its customers of the abnormalities experienced at the Byron and Braidwood plants. TB 03-6 also provides recommendations for plants to monitor their instrumentation to identify promptly any such abnormalities at their plants. The staff met with Westinghouse on September 26, 2003, to discuss efforts Westinghouse has taken to identify the cause of these abnormalities.

Westinghouse has not completed its root-cause evaluation of the problems, but currently believes that plant equipment near the instruments could have caused contamination in the signal, thus leading to incorrect readings by the flow meter. Westinghouse has also preliminarily concluded that this issue is limited to Byron and Braidwood. Based on current information, the staff does not believe that this issue poses an immediate safety concern. The staff is closely following this issue for Byron and Braidwood, as well as any implications on instrument installations for MUR power uprates.

SOURCE INFO
Source doc

[Source: "Monthly Status Report On The Licensing Activities And Regulatory Duties Of The United States Nuclear Regulatory Commission, September 2003", enclosure to Nils J. Diaz (Chairman, NRC), letter to Sen. George V. Voinovich (Chairman, Subcommittee on Clean Air, Climate Change and Nuclear Safety, Senate Committee on Environment and Public Works), November 25, 2003, ACN ML032900350]

IR 2003-006 BRAIDWOOD INSP REPORT 50000457-02

Accuracy Previously available

IR 50000456-2003006

.4 Potential Error in Reactor Thermal Power Calculation Due To Feedwater Flow Signal Noise

On August 31, 2003, the licensee reported to the NRC, via the Emergency Notification System, in accordance with 10 CFR 50.72, that Unit 2 had potentially exceeded its maximum licensed thermal power level of 3586.6 megawatts thermal, as stated in License Condition 2.C.(1), by up to 0.8 percent, on at least one occasion since a power uprate in 2001. Additional information was provided to the NRC on September 2, 2003, via the Emergency Notification System. The issue involved potential signal noise problems in the Advanced Measurement and Analysis Group (AMAG) ultrasonic feedwater flow detectors on both units. These detectors provide correction factors that are incorporated into the calorimetric thermal power calculations. The licensee reduced thermal power output by the appropriate amount and removed the AMAG correction factors from the calorimetric calculation. The licensee entered this issue into its corrective action system as CRs 173548 and 173819. The NRC was previously reviewing similar issues at the Byron Station as discussed in Inspection Report 50-454/03-02; 50-455/03-02, Section 4OA2.2. Since the NRC's review of this issue was

PUBLICLY AVAILABLE
IR 05000456/2003006

still ongoing, this issue is a URI (05000457/2003006-02).

.5 Potential Error in Reactor Thermal Power Calculation Due To Incorrect Heat Input Data

On September 3, 2003, the licensee reported to the NRC, via the Emergency Notification System, in accordance with 10 CFR 50.72, that both Unit 1 and Unit 2 had potentially exceeded their maximum licensed thermal power level limits of 3586.6 megawatts thermal, as stated in License Condition 2.C.(1), by up to 0.011 percent, on several occasions since power uprates were made to both units in 2001. This issue involved information provided by NASL-03-6 [Westinghouse Nuclear Safety Advisory Letter], in which errors were found in heat input values in the plant calorimetric calculation. The licensee reduced thermal power output by the appropriate amount and updated the calorimetric calculation to account for the errors. The licensee entered this issue into its corrective system as CR 173182. This issue involved a different cause, but a similar effect, as the one discussed above in Section 4OA3.4. Since the NRC's review of these issues were still ongoing, this issue is a URI (05000456/2003006-03; 05000457/2003006-03).

IR 2003008

PUBLICLY AVAILABLE
LER 05000456/2003-003-00

.1 (Closed) Licensee Event Report (LER) 05000456/2003-003-00 and Unresolved Items (URI) 05000456/2003006-03 and 05000457/2003006-03: Licensed Maximum Power Level Exceeded due to an Error in a Westinghouse Supplied Calorimetric Calculation Constant. LER TITLE

The inspectors reviewed the LER, related CRs, and other associated documents as listed in the Attachment at the end of this report. The inspectors also discussed the event with appropriate members of the licensee's engineering and operating staff.

This issue was previously described in Section 4OA3.5 of Inspection Report 05000456/2003006; 05000457/2003006. As stated in that report, the licensee identified that an error in the Westinghouse calculations used to determine net heat value for Braidwood Units 1 and 2 had resulted in both units potentially exceeding their licensed thermal power limits by 0.4 megawatts thermal (MWt). The net heat value was the amount of heat, seen at the steam generators, which was not supplied by the reactor. This value consisted of the heat supplied by the reactor coolant pumps, pressurizer heaters, and other minor heat removals and additions. The heat removed by reactor coolant pump seal leak off was non-conservatively omitted from the net heat calculation.

The licensee's corrective actions, as described in the LER, included reducing both units reactor power to 99.98 percent and notifying the NRC of the potential violation of License Condition 2.C(1), "Maximum Power Level." Additionally, the licensee verified that the additional 0.4 MWt was bounded by the existing design-bases calculations for reactor thermal power.

The inspectors determined that this issue was not a licensee performance deficiency and was therefore not considered a finding. However, in order to characterize the significance level of the license violation, the inspectors used the Significance Determination Process (SDP) and concluded that this issue was of very low safety significance (Green). Specifically, using the SDP Phase 1 Screening Worksheet of

IMC 0609, Appendix A, Attachment 1, the inspectors determined that the 0.4 MWt increase in Units 1 and 2 reactor power did not significantly challenge either the reactor coolant or fuel integrity barriers. The licensee entered this item into its corrective action system as CR 173182. The enforcement aspects of this issue are discussed in Section 4OA7.

.2 <u>(Closed) LER 05000457/2003-002-00 and URI 05000457/2003006-02</u>: Licensed Maximum Power Level Exceeded Due to Inaccuracies in Feedwater Ultrasonic Flow Measurements Caused by Signal Noise Contamination. LER TITLE

The inspectors reviewed the LER, related CRs, and other associated documents as listed in the Attachment at the end of this report. The inspectors also discussed the event with appropriate members of the licensee's engineering and operating staff.

This issue was previously described in Section 4OA3.4 of Inspection Report 05000456/2003006; 05000457/2003006. As stated in that report, the licensee used an ultrasonic flow measurement system to measure the time needed for feedwater disturbances to travel a known distance in the feedwater piping. The licensee identified that this system was incorrectly measuring feedwater flow owing to signal noise generated by feedwater flow pressure pulses. These pressure pulses were caused by the feedwater regulating valves, which were located upstream of the system. The pressure pulses and the resultant noise were not identified during the initial installation and subsequent testing of the ultrasonic system.

Already Publicly available

LER 05000457/2003-002-00

The licensee's corrective actions, as described in the LER, included reducing Unit 2 reactor power to below the licensed thermal power limit, removing the ultrasonic system from service, and notifying the NRC of the potential violation of License Condition 2.C(1), "Maximum Power Level." Additionally, the licensee was performing a technical review to determine how to reconfigure the ultrasonic system to reduce or remove the signal noise. This included a planned revision to station procedures to periodically check the ultrasonic system for signal noise and to remove the system from service if significant noise were found. The licensee also verified that a similar ultrasonic system installed on Unit 1 was operating correctly and that no overpower condition existed.

The inspectors determined that this issue was not a licensee performance deficiency and was, therefore, not a finding. However, in order to characterize the significance level of the license violation, the inspectors used the SDP and concluded that it was of very low safety significance (Green). Specifically, using the SDP Phase 1 Screening Worksheet of IMC 0609, Appendix A, Attachment 1, the inspectors determined that the 1.2 percent increase (as determined using the venturis to measure feedwater flow) in Unit 2 power did not significantly challenge either the reactor coolant or fuel integrity barriers. The inspectors' analysis also included the additional 0.4 MWt from the error in the Westinghouse calculation as described in Section 4AO3.1. The licensee entered this item into its corrective action system as CRs 173548 and 173819. The enforcement aspects of this issue are discussed in Section 4OA7.

IR 2004003

.4 <u>Potential Operation of Unit 1 Above the Licensed Thermal Power Limit</u>

Release Public

On March 1, 2004, the licensee reported via the Emergency Notification System that it had determined that Unit 1 had potentially exceeded its maximum licensed thermal power level of 3586.6 megawatts thermal, as stated in License Condition 2.C.(1), by up to 1.07 percent on at least one occasion between June 1999 and September 2003. The issue involved signal noise problems in the ultrasonic feedwater flow detectors. This was the same issue as previously reported and discussed in an Event Notification dated August 31, 2003, and updated on September 2, 2003, LER 05000457/2003-002-00, Inspection Report 05000456/2003006; 05000457/2003006, Section 4OA3.4, and Inspection Report 05000456/2003008; 05000457/2003008, Sections 4OA3.2 and 4OA7.

The new information in this notification was that Unit 1 may have been affected enough to have exceeded its licensed limit rather than only Unit 2 as previously reported. This was based on new testing at both the Braidwood and Byron stations that indicated that the feedwater flow error could have been greater than originally reported. Because of questions regarding the accuracy of the ultrasonic flow instrumentation, the licensee had removed them from service on both units in September 2003.

As previously discussed in Inspection Report 05000456/2003008; 05000457/2003008, Section 4OA3.2, the inspectors determined that the issue was not a licensee performance deficiency and was, therefore, not a finding. As discussed in that report, the inspectors determined that the potential overpower did not significantly challenge either the reactor coolant or fuel integrity barriers and was of very low safety significance. The potential overpower condition was within the bounds of the assumptions in the accident analysis in the UFSAR. The licensee entered the issue into its corrective action program as CR 205273 and intended to revise LER 05000457/2003-002 with the new information by March 31, 2004. The enforcement aspects of this issue are discussed in Section 4OA7 of this report.