WOLF CREEK
NUCLEAR OPERATING CORPORATION

Otto L. Maynard
Vice President Plant Operations

April 5, 1993
WO 93-0068

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

Subject: Docket No. 50-482: Licensee Event Report 93-001-00

Gentlemen:

The attached Licensee Event Report (LER) is being submitted as a voluntary report.

Very truly yours,

Otto L. Maynard

Otto L. Maynard
Vice President
Plant Operations

OLM/jan

Attachment

cc: W. D. Johnson (NRC), w/a
J. L. Milhoan (NRC), w/a
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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Wolf Creek Generating Station	0 5 0 0 0 4 8 2	1 OF 0 8

TITLE (4) Inadequate Technical Specifications Developed By Westinghouse Results In A Failure To Include All Time Delays In Technical Specification Response Time Testing

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 3	0 4	9 3	9 3	0 0 1	0 0	0 4	0 5	9 3		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 0 5 1

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

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)	X OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	Voluntary Report
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 AREA CODE	
Kevin J. Moles - Manager Regulatory Services	3 1 6	3 6 4 - 8 8 3 1

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)

YES (If yes, complete EXPECTED SUBMISSION DATE) | X NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On March 4, 1993, at approximately 1150 CST, it was determined that the plant was potentially in a condition outside its design basis since not all time delays associated with the Reactor Coolant Pump (RCP) Undervoltage Trip function were included in the response time testing of the RCP Undervoltage Reactor Trip Circuit. A subsequent evaluation concluded that although all testing required to fulfill all of the assumptions within the Safety Analysis were not performed, the plant never operated in a condition outside its design basis. However, it was determined by management that due to the potential generic significance of the issue a voluntary Licensee Event Report would be submitted.

The root cause of this condition was inadequate Technical Specifications developed by Westinghouse for Wolf Creek Generating Station implementation. Although surveillance testing was accomplished in accordance with the requirements of Technical Specifications, the testing was not performed to the design basis. The appropriate procedures will be revised to include the additional time delays by June 1, 1993. Also, the potential generic implications of this event will be brought to the attention of the Westinghouse Owners' Group for further consideration and review.

BASIS FOR VOLUNTARY REPORTABILITY

On March 4, 1993, at approximately 1150 CST, it was determined that the plant was potentially in a condition outside its design basis since not all time delays associated with the Reactor Coolant Pump (RCP) [AB-P] Undervoltage Trip function were included in the response time testing of the RCP Undervoltage Reactor Trip Circuit [JC-27]. A subsequent evaluation concluded that although all testing required to fulfill all of the assumptions within the Safety Analysis were not performed, the plant never operated in a condition outside its design basis. This conclusion was made based on a review of actual historical test data which indicated that sufficient margin existed between the Technical Specification response time limit and the actual channel response time to accommodate the additional response delay times that were not tested for. Therefore, there were no cases where Technical Specifications were violated and there were no safety concerns. However, it was determined by management that due to the potential generic significance of the issue a voluntary Licensee Event Report would be submitted.

PLANT CONDITIONS AT TIME OF DISCOVERY

Mode 1, Power Operation, at approximately 51 percent reactor power, reducing power in preparation for the sixth refueling outage.

Average Reactor Coolant System Temperature - 599 degrees Fahrenheit

Reactor Coolant System Pressure - 2237.5 pounds per square inch gauge

DESCRIPTION OF EVENT

Technical Specification 3.3.1 requires, in part, that Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 be operable with response times as defined in Table 3.3-2. Table 3.3-2 defines the response time for the RCP Undervoltage Trip function as being less than 1.5 seconds. The Technical Specification Bases for Technical Specification 3/4.3.1 states that "The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analysis."

Westinghouse Technical Bulletin NSD-TB-92-03, entitled "Undervoltage Trip Protection," was issued on May 26, 1992, to clarify apparent confusion at some plants concerning the Reactor Trip System [JC] Response Time accounted for in the plant Safety Analyses and listed in the plant's Technical Specifications for the RCP Undervoltage Trip function. The confusion was a result of the interpretation of the definition of response time in the plant Technical Specifications, wording in some plant's Bases section of the plant Technical Specifications, and the unique accounting of channel response time in the Safety Analyses for the Complete Loss of Flow accident scenario which credits the RCP Undervoltage Trip function. Specifically, not all time delays associated with this trip function are included in the response time testing of the RCP Undervoltage Reactor Trip Circuit.

The Westinghouse Technical Bulletin stated that the RCP Undervoltage Trip channel response time accounted for in the Safety Analysis for the Complete Loss of Flow Accident is unique in that non-hardware related delay times are included in the total analyzed time. Typically, Safety Analysis response times include hardware related delay times only. The RCP Undervoltage Trip channel response time is comprised of the Electro-Motive Force (EMF) delay from the time of the loss of bus voltage until the EMF generated by the bus loads has decayed to a value less than the undervoltage trip setpoint, the inherent undervoltage sensing circuitry time delay from the time the undervoltage trip setpoint is reached until an undervoltage reactor trip signal is generated, the intentional time delay set into the undervoltage relay actuation to prevent spurious reactor trips from momentary electrical power transients, and the time delay for the reactor trip breakers [AA-BKR] to open and the Rod Cluster Control Assembly (RCCA) grippers [AA-CPL] to release. These values are added together to obtain a total response time for the RCP Undervoltage Relay channels [JC-CHA]. Of these values, the EMF delay time and the RCCA gripper release delay time components are not determined during Technical Specification surveillance testing for the total response time of the RCP Undervoltage Reactor Trip Circuit.

The EMF delay time is the anticipated time it would take for the Undervoltage Relay to detect a loss of voltage when there is an instantaneous loss of bus voltage. When the supply voltage to the Reactor Coolant Pumps (RCP's) is lost instantaneously, the potential exists for the RCP's, or other loads on the bus, to generate an EMF back onto the bus during coastdown. The effect of this back EMF is to cause the voltage across the bus to decay at some finite rate instead of instantaneously. The Undervoltage Relay may not detect the instantaneous loss of voltage but may be influenced by the EMF being generated by the bus loads. This will cause a delay in the time it takes for the bus voltage to decay to a value below the Undervoltage Relay Trip setpoint. Based on Wolf Creek Generating Station (WCGS)

Technical Specification Definition 1.26, which states that "The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage," the monitored parameter has exceeded its trip setpoint, but the channel sensor (Undervoltage Relay) is unable to detect this due to the back EMF. Westinghouse assumes a generic EMF delay time of 250 milliseconds (mS).

The Westinghouse Technical Bulletin also interprets "loss of stationary gripper coil voltage" stated in Technical Specification Definition 1.26 to mean "rods are free to fall." However, WCGS interpreted the loss of stationary gripper coil voltage to mean that the Reactor Trip Breakers are open. This interpretation does not account for the time delay for the RCCA grippers to release. Westinghouse assumes a generic gripper release delay time of 150 mS.

Following receipt of Westinghouse Technical Bulletin NSD-TB-92-03, it was incorporated into the Industry Technical Information Program (ITIP) and was forwarded to the design engineering group for evaluation. A preliminary evaluation concluded that although the EMF delay time was not being surveillance tested, there was sufficient margin in the actual RCP Undervoltage Trip channel response time to allow for the additional time delay and still be able to meet the Technical Specification required response time of 1.5 seconds. Therefore, the final evaluation was given a routine priority for completion.

As the final evaluation neared completion it became apparent that the amount of margin was much tighter than was originally determined and it was indeterminate whether WCGS had always met the Technical Specification required response time of 1.5 seconds for the RCP Undervoltage Trip channel. On March 4, 1993, at approximately 1150 CST, a conservative decision was made to report this item as a condition potentially outside the design basis and at 1235 CST the Nuclear Regulatory Commission (NRC) Operations Center was notified pursuant to 10 CFR 50.72(b)(1)(ii)(B). However, a preliminary review of the most current Technical Specification Surveillance data indicated that the actual RCP Undervoltage Trip channel response time, as measured by previous surveillance tests, was within the required time of 1.5 seconds.

In an effort to determine the actual delay times associated with each response time test which are used to determine total response time for the RCP Undervoltage Trip channel, past surveillance procedures were reviewed to identify the worst case as found/as left values. The worst case total delay time that has existed at WCGS for these surveillance tests was determined to be 1.215 seconds, but these tests did not include the gripper release delay time and the EMF delay time. WCGS

specific gripper release times were determined from a review of 265 individual rod drop tests previously performed during surveillance test procedure STS RE-007, "Rod Drop Time Measurement." The worst case gripper release delay time has been 60 mS. A preliminary calculation determined the WCGS specific worst case EMF delay time to be 200 mS. Adding the additional time delays (60 mS and 200 mS) to the worst total delay time gives a new total delay time of 1.475 seconds. Therefore, since this is less than the 1.5 seconds allowed by Technical Specification Table 3.3-2, there has been no violation of the Technical Specification requirements. Also, since the surveillance testing that has previously been accomplished meets the requirements of the Technical Specification 1.26 definition for Reactor Trip System Response Time, Technical Specification Surveillance requirements were not violated.

An engineering evaluation conducted in support of the investigation concluded that even though allowances were not made for the EMF delay time and the gripper release delay time, actual plant historical test data indicated that sufficient margin existed between the Technical Specification limit and the actual channel response time to accommodate the excluded delay times. While the RCP Undervoltage reactor trip response time surveillance procedures have not previously taken into account the two additional delays, had they been included in the procedures, the plant would have met the required Technical Specification limit of 1.5 seconds. Therefore, the plant was never operated in a condition outside its design basis. Based on the results of this evaluation, a retraction phone call was made to the NRC Operations Center on March 18, 1993, at 1425 CST to notify them that the plant was never in a condition that was outside its design basis and that there was no basis for reportability pursuant to 10 CFR 50.72 or 10 CFR 50.73. However, it was determined by WCGS management that a voluntary Licensee Event Report would be submitted.

ROOT CAUSE

The root cause of this condition was inadequate Technical Specifications developed by Westinghouse for WCGS implementation. Technical Specification Definition 1.26, which states that "The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage," did not adequately account for the additional time delays identified in Westinghouse Technical Bulletin NSD-TB-92-03. Thus, the associated surveillance tests were not accordingly adequate. Although surveillance testing was accomplished in accordance with the requirements of Technical Specifications, the testing was not performed to the design basis.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

CORRECTIVE ACTIONS

A review was conducted to determine if the EMF delay time, the gripper release time, or similar analysis assumptions, were applicable to other Reactor Trip System Response Times (i.e., other than the RCP Undervoltage protection). This review concluded that the RCP Undervoltage Trip function was the only Reactor Trip System Response Time that appears in Table 3.3-2 of the Technical Specifications to be affected by the EMF delay time. However, the gripper release time is applicable to all Reactor Trip System Response Times listed in Table 3.3-2. Each Reactor Trip results in the deactivation of the stationary gripper coil by the removal of the coil voltage source, therefore, all trip channel response times have a gripper release delay time component. Therefore, response time surveillance tests for the trip systems listed in Table 3.3-2 must include an allowance for gripper release delay time.

A 80 mS gripper release delay time, 20 mS greater than the worst actual case found, will be included in the Reactor Trip System Response Times listed in Table 3.3-2 of the Technical Specifications by reducing the allowed reactor trip breaker response time from the current 167 mS to 85 mS. Therefore, both the trip breaker response time and gripper release delay time can be included in the time span currently allowed for reactor trip breaker response time. Historical data indicate that trip breaker response times fall within the proposed limit, so no adverse impact is expected from the change. Also, the EMF delay time of 200 mS will be included in the acceptance criteria of the RCP Undervoltage channel calibration procedures. Although the 200 mS cannot be verified by response time testing, the 200 mS will be incorporated into the acceptance criteria of the procedures. The Reactor Trip System Response Times listed in Table 3.3-2 and required for the sixth refueling outage will be verified to be within the Technical Specification limit, including the EMF delay time and the gripper release delay time, prior to startup from the sixth refueling outage. The applicable procedures will be revised prior to June 1, 1993.

Surveillance test procedure STS RE-007, "Rod Drop Time Measurement," will also be revised to determine the gripper release delay time. Previously, the gripper release delay time was not specifically determined during the performance of the procedure, but was included in the total reactor trip breaker response time. Surveillance test procedure STS RE-007 will be revised to determine the gripper release delay time, with an acceptance criteria of 80 mS prior to its performance during the sixth refueling outage.

GENERIC IMPLICATIONS

Safety Analysis assumptions provide the basis for WCGS Technical Specification limits and surveillance procedures. Currently, when a safety analysis assumption is changed, appropriate WCGS internal documents are generated to communicate the change to the Operations group. Conversely, when the Operations group requires a change which could impact the safety analysis, appropriate internal documents are transmitted to request approval of the change by the Nuclear Analysis group. Therefore, sufficient communication exists between the Operations group and Nuclear Analysis to allow appropriate changes in plant surveillance procedures or other plant documents. To capture cycle-to-cycle changes, a Reload Design Initialization Checklist (RDIC) is routed throughout the Engineering group and the Operations group to request anticipated plant and operational changes. This is done to ensure that a bounding analysis may be performed in the reload design process. It can be concluded that the existing processes for ensuring the underlying assumptions from the applicable safety analyses are appropriately included in plant documents are sufficient and adequate to handle future problems that may arise, once they are known to exist.

During the process of accepting responsibility from Westinghouse for the WCGS Safety Analyses, reviews were conducted of the underlying assumptions for plant impact. To ensure the correctness and agreement of the safety analysis assumptions in relation to plant design, several engineering audits were performed during 1985 and 1986 on the Westinghouse analysis. To facilitate the audits, members of the WCGS and Callaway Nuclear Analysis groups, assisted by contractors, traveled to the Westinghouse offices to review individual calculations. These audits looked for discrepancies between the Westinghouse calculation, the Final Safety Analysis Report (FSAR) write-up, and plant design. The audit results discovered several instances where discrepancies existed between the calculation and the FSAR. All items were resolved, many with revisions to the FSAR, to the satisfaction of WCGS personnel. The audit results also indicated that none of the discrepancies involved unreviewed safety questions and that the appropriate plant information had been included in the safety analysis and vice versa.

Reviews regarding the underlying safety analysis assumptions for plant impact have been performed for the discrepancies which were discovered during the audits between what was reported in the FSAR and what was actually used in the safety analysis. There have been cases where it has been found that either the safety analysis has not correctly reflected plant design, or cases where plant documents were not written to appropriately reflect the underlying safety analysis assumptions. When these situations have occurred, evaluations have been performed to assess plant impact and appropriate documentation changes and reporting

procedures have been followed to ensure that the condition was corrected. It has been concluded that a sample review of any additional safety analysis and their assumptions would not be a worthwhile expenditure of resources. This conclusion was drawn since this item did not present any safety problems or violate any NRC requirements, and any other items found from doing such a sample review would likely result in the same conclusions for those items upon evaluation.

The potential generic implications of this event will be brought to the attention of the Westinghouse Owners' Group (WOG) for further consideration and review.

SAFETY ANALYSIS

As previously discussed, an evaluation conducted by Nuclear Analysis concluded that although all testing required to fulfill all of the assumptions within the Safety Analysis were not performed, the plant was never operated in a condition outside its design basis. This conclusion was made based on a review of actual WCGS historical surveillance test data which indicated that sufficient margin existed between the Technical Specification response time limit and the actual channel response time to accommodate the additional delay times that were not previously tested for. Therefore, no safety concerns existed and the condition did not pose a threat to the health and safety of the public or plant safety.

PREVIOUS SIMILAR OCCURRENCES

Licensee Event Reports (LER) 85-080-00, 86-003-00, 86-067-00, 87-060-00, 89-006-00, 91-003-00, 91-007-00, 92-005-00, and 92-013-00 discuss previous similar occurrences in which Technical Specification Surveillance Requirements were not incorporated into surveillance procedures. None of these events involved a failure to include the underlying assumptions of the Safety Analysis in the surveillance test procedures except LER 87-060-00. LER 87-060-00 describes an event in which it was identified that the P-4 Turbine Trip on Reactor Trip function had not been incorporated into the surveillance procedures although it had been assumed in the safety analysis.