

MAY 14 1987

*Correction to Amdt. 22
to NPF-30*

Docket No. 50-483

Mr. Donald F. Schnell
Vice President - Nuclear
Union Electric Company
Post Office Box 149
St. Louis, Missouri 63166

Dear Mr. Schnell:

On May 4, 1987, the Commission issued Amendment No. 22 to Facility Operating License No. NPF-30 for the Callaway Plant, Unit No. 1. The amendment consisted of changes to the Technical Specifications in response to your application dated April 16, 1987.

In the process of printing the unchanged back-up page B 3/4 3-1, four lines were inadvertently transferred from the top of page B 3/4 3-2 to the bottom of page B 3/4 3-1. The correction has been made to page B 3/4 3-1 and a corrected copy is enclosed.

Please accept our apologies for any inconvenience this error may have caused you.

Sincerely,

/s/

Thomas W. Alexion, Project Manager
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As Stated

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Callaway Plant
Unit No. 1

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3/4.3 INSTRUMENTATION

BASES

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated action and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

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 and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service times for the Reactor Protection Instrumentation System," supplements to that report, and the NRC's Safety Evaluation dated February 21, 1985. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 3.3-1, $Z + R + S < TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and

INSTRUMENTATION

BASES

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for the actuation. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the Reactor Trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements or (2) utilizing replacement sensors with certified response times.

Engineered Safety Features response time specified in Table 3.3-5 which include sequential operation of the RWST and VCT valves (Notes 3 and 4) are based on values assumed in the non-LOCA safety analyses. These analyses take credit for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction valves are closed following opening of the RWST charging pump suction valves. When the sequential operation of the RWST and VCT valves is not included in the response times (Note 7), the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response time specified in Table 3.3-5 will assure that the assumptions used for the LOCA and non-LOCA analyses with respect to operation of the VCT and RWST valves are valid.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to