ULNRC-1493
Attachment 1

PROPOSED TECHNICAL SPECIFICATION CHANGE

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TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
e. Containment Purge Isolation	N.A.
f. Steam Line Isolation	N.A.
g. Feedwater Isolation	N.A.
h. Auxiliary Feedwater	N.A.
i. Essential Service Water	N.A.
j. Containment Cooling	N.A.
k. Control Room Isolation	N.A.
l. Reactor Trip	N.A.
m. Emergency Diesel Generators	N.A.
n. Component Cooling Water	N.A.
o. Turbine Trip	N.A.
2. Containment Pressure-High-1	
a. Safety Injection (ECCS)	≤ 29~~[1]~~[7]/~~12~~ 27[4]
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 2[5]
3) Phase "A" Isolation	≤ 1.5[5]
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	≤ 60[1]
6) Containment Cooling	≤ 60[1]
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	≤ 14[6]
9) Turbine Trip	N.A.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq 29^{\cancel{(1)}\,(7)}/\cancel{12}\,27^{(4)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 2^{(5)}$
3) Phase "A" Isolation	$\leq 2^{(5)}$
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	$\leq 60^{(1)}$
6) Containment Cooling	$\leq 60^{(1)}$
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	$\leq 14^{(6)}$
9) Turbine Trip	N.A.
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq \cancel{24}\,37^{(3)}/\cancel{12}\,27^{(4)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 2^{(5)}$
3) Phase "A" Isolation	$\leq 2^{(5)}$
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	$\leq 60^{(1)}$
6) Containment Cooling	$\leq 60^{(1)}$
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	$\leq 14^{(6)}$
9) Turbine Trip	N.A.
b. Steam Line Isolation	$\leq 2^{(5)}$

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
12. Auxiliary Feedwater Pump Suction Pressure-Low	
Transfer to Essential Service Water	N.A.
13. RWST Level-Low-Low Coincident with Safety Injection	
Automatic Switchover to Containment Sump	$\leq$ 60
14. Loss of Power	
a. 4 kV Bus Undervoltage-Loss of Voltage	$\leq$ 14
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	$\leq$ 144
15. Phase "A" Isolation	
a. Control Room Isolation	N.A.
b. Containment Purge Isolation	$\leq 2^{(5)}$

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting delay not included. Offsite power available.

(3) Diesel generator starting and sequence loading delay included. RHR pumps not included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(4) Diesel generator starting and sequence loading delays not included. Offsite power available. RHR pumps not included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(5) Does not include valve closure time.

(6) Includes time for diesel to reach full speed.

(7) Diesel generator starting and sequence loading delays included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is not included. Response time assumes only opening of RWST valves.

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 statisitical 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.

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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 those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trips, (3) Feedwater System isolates, (4) the emergency diesel generators start, (5) containment spray pumps start and automatic valves position, (6) containment isolates, (7) steam lines isolate, (8) Turbine trips, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment cooling fans start and automatic valves position, (11) essential service water pumps start and automatic valves position, and (12) isolate normal control room ventilation and start Emergency Ventilation System.

INSERT A

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.

SAFETY EVALUATION

Reference: 1. ULNRC-1207 dated 11/15/85

This safety evaluation is in support of a license amendment request to revise Technical Specification Table 3.3-5 to increase the ESF response times for Items: 2.a. (Containment Pressure-High-1, SI); 3.a. (Pressurizer Pressure-Low, SI); and 4.a. (Steam Line Pressure-Low, SI). These changes are contained in Attachment 1.

BACKGROUND

In the normal configuration of the Chemical and Volume Control System (CVCS), the charging pumps take suction from the Volume Control Tank (VCT). When a Safety Injection (SI) signal is generated from the protection logic, a signal is sent to start the high-head charging pumps and to begin opening the Refueling Water Storage Tank isolation valves, in order to align the borated water source for delivery to the RCS. Once the RWST isolation valves have repositioned and are indicated fully opened, the isolation valves on the VCT will begin to close. This sequential valve stroke time can be as long as 25 seconds. Since the VCT is pressurized, it will be the source of the SI flow until the isolation valves are closed. This affects the time assumed at which the 2000 ppm borated water in the RWST is available to the suction of the charging pumps.

The FSAR Steam Line Break analysis (Reference 1) which supports the current Technical Specifications (Table 3.3-5) assumes the following delays for delivery of borated water to the RCS:

1. SI signal generation (2 seconds)

2. Diesel start-including time to come up to speed (12 seconds)

3. Valve stroke times and pumps to full speed (10 seconds)

This assumes, however, that the VCT and RWST isolation valves stroke simultaneously rather than sequentially. The valve interlock logic increases the delay time for the availability of borated water by 15 seconds (conservatively) to 27 seconds with offsite power and 39 seconds without offsite power. The only non-LOCA transient impacted by the increased time delay is the steam line break event. No other Chapter 15 transient relies on short-term boration from the RWST to mitigate the event.

EVALUATION

Based on the current steam line break analysis for the Callaway plant and sensitivities performed for other plants, the additional time delay is acceptable. Specifically:

1) The additional delay in the availability of borated water occurs early in the steam line break transient when RCS pressures are relatively high and SI flowrates are relatively small due to head vs. SI flow characteristics.

2) Previous sensitivities have shown that delays of this magnitude result in small changes in the analysis results. A comparison of cases with and without the additional SIS delay showed, over the limiting portion of the transient, maximum differences of 0.2% in power, 0.6 degrees F in temperature, and 10 psi in RCS pressure. A Callaway specific review of the steam line break analysis demonstrated that there is sufficient margin available in the analysis such that the conclusions presented in Reference 1 remain valid.

3) The analysis assumes only one centrifugal charging pump is available. However, at the pressures characteristic of a steam line break, the centrifugal charging pump and safety injection pump of a given train would be available to deliver a significantly greater flowrate of borated water to the RCS.

From analyses performed for other Westinghouse plants, it has been shown that SI boron concentration reduction has little effect on the steam line break mass/energy release analysis inside containment. Since the additional time delay is a small perturbation compared to a large change in the available boron concentration, there will be negligible impact on the steam line break mass/energy release inside containment analysis.

Sensitivities performed for the steam line break superheated mass/energy release outside containment analysis show that the results are not sensitive to large changes in SI flow (Reference WCAP-10961, Rev. 1). The additional time delay is a small perturbation compared to a large change in total SI flow; therefore, it is concluded that the impact on the Callaway superheated mass/energy releases outside containment is insignificant.

In the case of a Loss of Coolant Accident, the immediate safety function of SI is to supply water to the RCS, whether borated or not. The time at which water (from either the VCT or the RWST) is available to the suction of the high-head charging pumps is not affected. Thus, for those SI actuation signals that are only intended to provide protection against a LOCA, this additional delay is not required since boron is only required for maintaining subcriticality in the long term following a LOCA.

Effect on Design Basis Accident Analysis

A reference steam line break event for a four loop, 17 x 17 optimized fuel, PWR power plant was used to evaluate the sensitivity to SI flow. It was found that the difference in core boron concentration, peak return to power, RCS temperature, RCS pressure and DNBR were minimal with a 15 second delay in SI flow. The Callaway specific analysis was checked to ensure that sufficient margin existed.

Potential for Creation of an Unanalyzed Accident

There are no new failure modes associated with this proposed change since no design changes have been made.

No new accident is created because the same equipment is assumed to perform in the same manner as before. Only the testing of the timing of the delivery of borated injection flow is affected. This can be adequately modeled in the current safety analysis.

Effect on the Margin of Safety

There is no impact on the consequences on protective boundaries. All acceptance criteria in Reference 1 are still met.

The proposed change is intended to bring the Technical Specification surveillance in line with the basis. The basis is to mitigate a steam line break which requires injection of borated water into the RCS. The present Technical Specification surveillance ensures flow initiated to the core but did not test the time to provide borated water. The proposed change will increase the time to initiate borated water flow to the core by 15 seconds. With the additional 15 seconds delay in supplying borated water to the core, the DNB design basis is still met, and the conclusions in Reference 1 remain valid. Therefore, the change does not reduce the margin of safety as specified in the basis of any Technical Specification.

Summary & Conclusions

The proposed change in the ESF response times for Containment Pressure-High-1, Low Pressurizer Pressure and Low Steam Line Pressure in Technical Specification Table 3.3-5, Items 2.a, 3.a and 4.a to incorporate an increase of 15 seconds is acceptable. Evaluation of the impact on the Callaway safety analysis licensing basis demonstrates that the conclusions in Reference 1 remain valid.

Based on the foregoing assessment, the change proposed herein is considered safe and does not represent an unreviewed safety question as defined in 10CFR50.59 since is does not:

1. Increase the frequency of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report;

2. Create the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report;

3. Reduce the margin of safety as defined in the basis for any technical specification.

This amendment request would not adversely affect or endanger the health and safety of the general public and does not involve an unreviewed safety question.

SIGNIFICANT HAZARD EVALUATION

This significant hazard evaluation is in support of a license amendment request to revise Technical Specification Table 3.3-5 to increase the Engineered Safety Features (ESF) response times for Items: 2.a. (Containment Pressure-High-1, SI); 3.a. (Pressurizer Pressure-Low, SI); and 4.a. (Steam Line Pressure-Low, SI).

In accordance with 10CFR50.92, Union Electric Company has reviewed the proposed changes and has concluded they do not involve a significant hazards consideration. The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised, a conclusion which is supported by our determinations made pursuant to 10CFR50.59. The proposed change does not involve a significant hazards consideration because the change would not:

1) Involve a significant increase in the probability or consequences of accident previously evaluated. An increase in the acceptance criterion for the ESF response time is acceptable since the evaluation of the impact of the increased delay on the steam line break event demonstrated that the DNB design basis is still met. The conclusions presented in the ULNRC-1207 dated November 15, 1985 remain valid.

2) Create the possibility of a new or different kind of accident from any previously evaluated. There are no new failure modes associated with this proposed change, as no design changes have been made. No new accident is created because the same equipment is assumed to perform in the same manner as before. Therefore, an increase in the ESF response times for high containment pressure, low steam line pressure, and low steam line pressure does not create the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report.

3) Involve a significant reduction in a margin of safety. The proposed change is intended to bring the Technical Specification surveillance in line with the basis. As stated before, there is no impact on the consequences on protective boundaries, and all acceptance criteria in the analysis of record, submitted by ULNRC-1207 dated November 15, 1985, are still met. Therefore, the safety limits will still be met.

Moreover, the Commission has provided guidance concerning the application of standards in 10CFR50.92 by providing certain examples (March 6, 1986, FR7751) of amendments that are considered not likely to involve significant hazards consideration. Although the proposed change herein is not enveloped by a specific example, the proposed change would not involve a significant increase in the probability or consequences of an accident previously analyzed. The results of the safety evaluation show that there is sufficient margin available in the analysis such that the conclusions presented in ULNRC-1207 dated November 15, 1985 remain valid.

Application Fee

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