

March 16, 1994

Docket Nos. 50-498
and 50-499

Mr. William T. Cottle
Group Vice-President, Nuclear
Houston Lighting & Power Company
South Texas Project Electric
Generating Station
Post Office Box 289
Wadsworth, Texas 77483

Dear Mr. Cottle:

SUBJECT: EVALUATION OF REVISION TO TECHNICAL SPECIFICATION BASES FOR FACILITY
OPERATING LICENSE NOS. NPF-76 AND NPF-80 - SOUTH TEXAS PROJECT,
UNITS 1 AND 2

By letter dated February 25, 1994, you submitted revisions to Technical
Specification Bases 3/4.3 and 3/4.5. The revised Bases reflect changes to the
technical specifications that were incorporated by Amendment Nos. 59 and 47 to
the licenses. These amendments were issued by letter dated February 17, 1994.
The staff has reviewed the proposed revisions to the Bases and has determined
that they are consistent with the amendments and are therefore acceptable.
Enclosed are the affected pages with their overleaf pages to maintain document
completeness.

Sincerely,

Original Signed By
Lawrence E. Kokajko, Senior Project Manager
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Division of Reactor Projects III/IV/V
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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 South Texas Project probabilistic safety assessment (PSA). Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System 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 2.2-1, $Z + R + S \leq TA$, 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 rack drift and the accuracy of the measurement. TA or Total Allowance

INSTRUMENTATION

BASES

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

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 the 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. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

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.

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, (2) Reactor trip, (3) feedwater isolation, (4) startup of the standby diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) reactor containment fan coolers start, (11) essential cooling water pumps start and automatic valves position, (12) Control Room Ventilation Systems start, and (13) component cooling water pumps start and automatic valves position.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through three cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume represent a spread about an average value used in the safety analysis and have been demonstrated by sensitivity studies to vary the peak clad temperature by less than 20°F. The limit on accumulator pressure ensures that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be opened within 12 hours, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of three independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Each subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. One ECCS is assumed to discharge completely through the postulated break in the RCS loop. Thus, three trains are required to satisfy the single failure criterion. Note that the centrifugal charging pumps are not part of ECCS and that the RHR pumps are not used in the injection phase of the ECCS. Each ECCS subsystem and the RHR pumps and heat exchanges provide long-term core cooling capability in the recirculation mode during the accident recovery period.

When the RCS temperature is below 350°F, the ECCS requirements are balanced between the limitations imposed by the low temperature overpressure protection and the requirements necessary to mitigate the consequences of a LOCA below 350°F. At these temperatures, single failure considerations are not required because of the stable reactivity condition of the reactor and the limited core cooling requirements. Only a single Low Head Safety Injection pump is required to mitigate the effects of a large-break LOCA in this mode. However, two are

EMERGENCY CORE COOLING SYSTEMS

BASES

ECCS SUBSYSTEMS (Continued)

provided to accommodate the possibility that the break occurs in a loop containing one of the Low Head pumps. Low Head Safety Injection pumps are not required inoperable below 350°F because their shutoff head is too low to impact the low temperature overpressure protection limits.

Below 200°F (MODE 5) no ECCS pumps are required, so the High Head Safety Injection pumps are locked out to prevent cold overpressure.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that, at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for flow testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA.

3/4.5.4 (This specification number is not used)

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA or a steamline break. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, (2) the reactor will remain subcritical in the cold condition (68°F to 212°F) following a small break LOCA assuming complete mixing of the RWST, RCS, Spray Additive Tank, Containment Spray System and ECCS water volumes with all control rods inserted except the most reactive control rod assembly (ARI-1), (3) the reactor will remain subcritical in cold condition following a large break LOCA (break flow area > 3.0 ft²) assuming complete mixing of the RWST, RCS, Spray Additive Tank, Containment Spray System and ECCS water volumes and other sources of water that may eventually reside in the sump post-LOCA with all control rods assumed to be out (ARO), and (4) long term subcriticality following a steamline break assuming ARI-1 and preclude fuel failure.

The maximum allowable value for the RWST boron concentration forms the basis for determining the time (post-LOCA) at which operator action is required to switch over the ECCS to hot leg recirculation in order to avoid precipitation of the soluble boron.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.5 and 10.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.