Nuclear Operating Company
South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

February 12, 2002
NOC-AE-02001260
File No.: G20.02.02
G21.02.02
10CFR50.59
STI: 31402426

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Technical Specification Bases Change

South Texas Project Technical Specification Bases pages B 3/4 5-2, B 3/4 6-1 and B 3/4 6-1a are attached for your information and updating of the NRC copy of the Technical Specification Bases. These changes are enhancements to the Technical Specification Bases with information regarding:

- the operability of a LHSI train in Mode 4 with RHR in service, and
- an inoperable air lock interlock mechanism

If there are any questions, please contact A.W. Harrison at (361) 972-7298 or me at (361) 972-7136.

Scott M. Head
[signature]
Manager, Licensing

mkj

Attachment: Revised Technical Specification Bases Pages B 3/4 5-2, B 3/4 6-1 and B 3/4 6-1a

A001

O:\QUALITY_AND_LICENSING\TSB\AMENDMENT\02\NOC-AE-02001260 cc:

Ellis W. Merschoff
Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, Texas 76011-8064

Mohan C. Thadani
Project Manager
U. S. Nuclear Regulatory Commission
1 White Flint North, Mail Stop: O-7D1
11555 Rockville Place
Rockville, MD 20852-2738

Cornelius F. O'Keefe
U. S. Nuclear Regulatory Commission
P. O. Box 289, Mail Code MN116
Wadsworth, TX 77483

A. H. Gutterman, Esquire
Morgan, Lewis & Bockius
1111 Pennsylvania Avenue, NW
Washington, DC 20004

M. T. Hardt/W. C. Gunst
City Public Service
P. O. Box 1771
San Antonio, TX 78296

A. Ramirez/C. M. Canady
City of Austin
Electric Utility Department
721 Barton Springs Road
Austin, TX 78704

Jon C. Wood
Matthews & Branscomb
112 East Pecan, Suite 1100
San Antonio, Texas 78205-3692

Institute of Nuclear Power
Operations - Records Center
700 Galleria Parkway
Atlanta, GA 30339-5957

Richard A. Ratliff
Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, TX 78756-3189

R. L. Balcom/D. G. Tees
Reliant Energy, Inc.
P. O. Box 1700
Houston, TX 77251

C. A. Johnson/A. C. Bakken, III
AEP - Central Power and Light Company
P. O. Box 289, Mail Code: N5022
Wadsworth, TX 77483

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

# ATTACHMENT

# REVISED BASES PAGES

3/4.5.1 ACCUMULATORS (Continued)

Verification every 31 days that power is removed from each accumulator isolation valve operator when the pressurizer pressure is ≥1000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only one accumulator would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated isolation valves when pressurizer pressure is < 1000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns.

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 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 to be rendered inoperable below 350°F because their shutoff head is too low to impact the low temperature overpressure protection limits.

In Mode 4 with an RHR train in service, an LHSI train is considered operable if the only impact to its operability is the associated RHR train in service (i.e., heat exchanger bypass valves not fully closed, heat exchanger outlet valve not fully open).

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

## 3/4.6 CONTAINMENT SYSTEMS

## BASES

### 3/4.6.1 PRIMARY CONTAINMENT

#### 3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guidelines values of 10 CFR Part 100 during accident conditions.

#### 3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, $P_a$ (41.2 psig). As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 $L_a$ before returning the Unit to service following performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the requirements of 10 CFR Part 50, Appendix J, Option B, and in accordance with the Containment Leakage Rate Testing Program.

#### 3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests. The surveillance testing for measuring leakage rates is consistent with the requirements of 10 CFR Part 50, Appendix J, Option B and in accordance with the Containment Leakage Rate Testing Program.

Because both the inner and outer doors of an air lock are designed to withstand the maximum expected post-accident containment pressure, closure of either door will support containment operability. The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Therefore, an inoperable interlock mechanism should be considered the same as an inoperable air lock door.

## 3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.5 psig, and (2) the containment peak pressure does not exceed the design pressure of 56.5 psig during LOCA or steam line break conditions.

The maximum peak pressure expected to be obtained from LOCA or steam line break event is 41.2 psig ($P_a$). The limit of 0.3 psig for initial positive containment pressure will limit the total pressure to 41.2 psig, which is less than design pressure and is consistent with the safety analyses.