5.5 Programs and Manuals (continued)

5.5.13 Diesel Fuel Oil Testing Program (continued)

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An API gravity or an absolute specific gravity within limits,

2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. Water and sediment within limits when tested in accordance with ASTM D1796;

b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the stored fuel oil is ≤ 10 mg/l when tested every 92 days in accordance with ASTM D-2276, Method A-2 or A-3.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license; or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

(continued)

5.5 Programs and Manuals (continued)

5.5.14 Technical Specifications (TS) Bases Control Program (continued)

d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

(continued)

5.5 Programs and Manuals (continued)

5.5.15 Safety Function Determination Program (continued)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.0 psig. The containment design pressure is 60 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1 % of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

(continued)

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program (continued)

b. Air lock testing acceptance criteria are:

1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq$14.5 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Shutdown Margin - Reactor Trip Breakers Open for Specification 3.1.1.

2. Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.2.

3. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.4.

4. Boron Dilution Alarm System for Specification 3.3.12.

5. CEA Alignment for Specification 3.1.5.

6. Regulating CEA Insertion Limits for Specification 3.1.7.

7. Part Length CEA Insertion Limits for Specification 3.1.8.

8. Linear Heat Rate for Specification 3.2.1.

9. Azimuthal Power Tilt - T_q for Specification 3.2.3.

10. DNBR for Specification 3.2.4.

11. Axial Shape Index for Specification 3.2.5.

12. Boron Concentration (Mode 6) for Specification 3.9.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

----------------------NOTE------------------------
The COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

1. "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A, (Methodology for Specification 3.1.7, Regulating CEA Insertion Limits).

2. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, [Methodology for Specifications 3.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient BOL and EOL limits; 3.1.7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

3. "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR System 80, Docket No. STN 50-470, "NUREG-0852 (November 1981), Supplements No. 1 (March 1983), No. 2 (September 1983), No. 3 (December 1987) [Methodology for Specifications 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient BOL and EOL limits; 3.3.12, Boron Dilution Alarm System; 3.1.5, CEA Alignment; 3.1.7, Regulating CEA Insertion Limits; 3.1.8, Part Length CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt - T_q].

4. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, (Methodology for Specification 3.2.4, DNBR and 3.2.5 Axial Shape Index).

5. "Calculative Methods for the CE Large Break LOCA Evaluation Model," CENPD-132, (Methodology for Specification 3.2.1, Linear Heat Rate).

6. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, (Methodology for Specification 3.2.1, Linear Heat Rate).

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

7. Letter: O.D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 5.6.5.b.6.

8. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 5.6.5.b.6.

9. "Fuel Rod Maximum Allowable Pressure," CEN-372-P-A, (Methodology for Specification 3.2.1, Linear Heat Rate).

10. Letter: A. C. Thadani (NRC) to A. E. Scherer (CE), dated April 10, 1990, ("Acceptance for Reference CE Topical Report CEN-372-P"). NRC approval for 5.6.5.b.9.

11. "Arizona Public Service Company PWR Reactor Physics Methodology Using CASMO-4/SIMULATE-3," [Methodology for Specifications 3.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.1.7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

12. "Technical Manual for the CENTS Code," CE-NPD 282-P-A, Volumes 1-3, [Methodology for Specifications 3.1.2, Shutdown Margin-Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.1.5, CEA Alignment; 3.1.7, Regulating CEA Insertion Limits; 3.1.8, Part Length CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt- T_q].

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

5.6 Reporting Requirements (continued)

5.6.6 PAM Report

When a report is required by Condition B or G of LCO 3.3.10, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged and/or repaired in each steam generator shall be reported to the Commission in a Special Report.

The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report within 12 months following completion of the inspection. This Special Report shall include:

a. Number and extent of tubes inspected.

b. Location and percent of wall-thickness penetration for each indication of an imperfection.

c. Identification of tubes plugged and/or repaired.

Results of steam generator tube and sleeve inspections which fall into Category C-3 shall be reported in a Special Report to the Commission within 30 days and prior to resumption of plant operation and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.