**Don E. Grissette**
Vice President

**Southern Nuclear**
**Operating Company, Inc.**
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201

Tel 205.992.6474
Fax 205.992.0341

SOUTHERN COMPANY
*Energy to Serve Your World*℠

November 10, 2004

Docket Nos.: 50-424
50-425

NL-04-2205

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

Vogtle Electric Generating Plant
Changes to Technical Specification Bases

Ladies and Gentlemen:

The Vogtle Electric Generating Plant (VEGP) Unit 1 and Unit 2 Technical Specifications, section 5.5.14, Technical Specifications (TS) Bases Control Program, provide for changes to the Bases without prior NRC approval. In addition, TS section 5.5.14 requires that Bases changes made without prior NRC approval be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). Pursuant to TS section 5.5.14, Southern Nuclear Operating Company hereby submits Bases changes made to the VEGP TS Bases under the provisions of TS section 5.5.14. This submittal reflects changes since November 17, 2002 through May 17, 2004.

This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

Don E. Grissette

DEG/DWM/daj

Enclosure: Bases Changes

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. W. F. Kitchens, General Manager – Plant Vogtle
RType: CVC7000

U. S. Nuclear Regulatory Commission
Dr. W. D. Travers, Regional Administrator
Mr. C. Gratton, NRR Project Manager – Vogtle
Mr. G. J. McCoy, Senior Resident Inspector – Vogtle

A001

Enclosure
Vogtle Electric Generating Plant
Bases Changes

| | |
|---|---|
| B 2.1.1-7 | Rev. 1-6/03 |
| B 3.1.4-5 | Rev. 1-8/03 |
| B 3.1.4-10 | Rev. 1-8/03 |
| B 3.3.1-1 | Rev. 1-9/03 |
| B 3.3.1-19 | Rev. 3-6/03 |
| B 3.3.1-21 | Rev. 2-6/03 |
| B 3.3.1-22 | Rev. 2-6/03 |
| B 3.3.1-31 | Rev. 1-9/03 |
| B 3.3.1-55 | Rev. 1-4/04 |
| B 3.3.1-56 | Rev. 1-4/04 |
| B 3.3.1-57 | Rev. 1-4/04 |
| B 3.3.1-58 | Rev. 1-4/04 |
| B 3.3.1-58a | Rev. 0-4/04 |
| B 3.3.1-58b | Rev. 0-4/04 |
| B 3.3.1-59 | Rev. 1-4/04 |
| B 3.3.1-60 | Rev. 2-4/04 |
| B 3.3.1-61 | Rev. 2-4/04 |
| B 3.3.1-62 | Rev. 3-4/04 |
| B 3.3.1-63 | Rev. 4-4/04 |
| B 3.3.1-64 | Rev. 3-4/04 |
| B 3.3.1-65 | Rev. 6-4/04 |
| B 3.3.1-66 | Rev. 2-4/04 |
| B 3.3.1-67 | Rev. 0-4/04 |
| B 3.3.2-46 | Rev. 1-12/03 |
| B 3.3.3-12 | Rev. 1-6/03 |
| B 3.4.13-5 | Rev. 1-01/03 |
| B 3.5.1-6 | Rev. 1-11/03 |
| B 3.5.1-8 | Rev. 2-11/03 |
| B 3.5.1-9 | Rev. 2-11/03 |

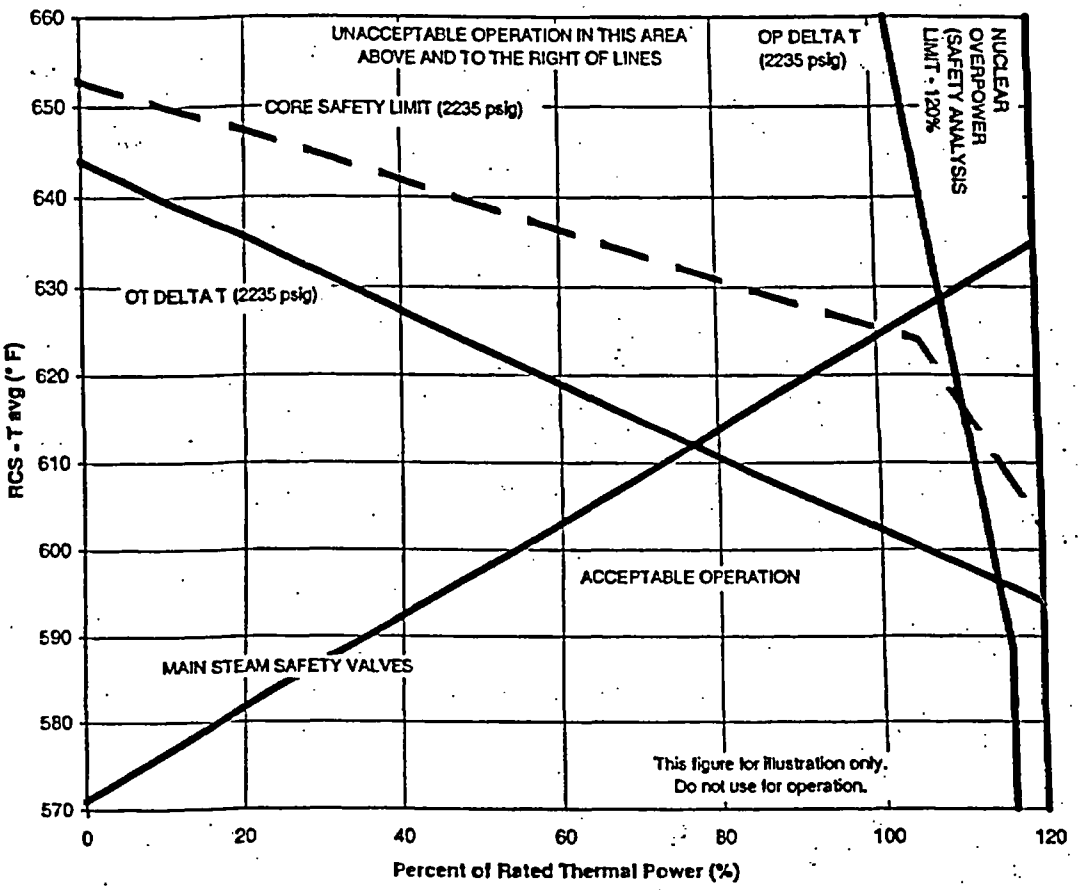


Figure B 2.1.1-1 (page 1 of 1)
REACTOR CORE SAFETY LIMITS VS. BOUNDARY OF PROTECTION

BASES (continued)

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements (i.e., trippability) are separate from the alignment requirements which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment.

The rod OPERABILITY (i.e., trippability) requirement is satisfied provided that the rod will fully insert in the required rod drop time assumed in the safety analyses. Rod control malfunctions that result in the inability to move a rod (e.g., rod lift coil failures), but that do not impact trippability, do not result in rod inoperability. However, where rod(s) are not moving, the rod(s) must be considered untrippable unless there is verification that a rod control system failure is preventing rod motion. If the rod control system is demanding motion properly and no motion occurs, the rod is considered untrippable (i.e., inoperable).

The requirement to maintain the rod alignment to within plus or minus 12 steps of their group step counter demand position is conservative. The safety analysis assumes a total misalignment from fully withdrawn to fully inserted. When required, movable incore detectors may be used to determine rod position and verify the rod alignment requirement of this LCO is met.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which a self-sustaining chain reaction ($K_{eff} \geq 1$) occurs, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are fully inserted and the reactor is shut down, with no self-sustaining chain reaction. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. If the rod position deviation monitor is inoperable, a Frequency of 4 hours accomplishes the same goal. The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

SR 3.1.4.2

Exercising each individual control rod every 92 days provides confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.4.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

SR 3.1.4.3

Verification of rod drop times from the physical fully withdrawn position allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect

(continued)

# B 3.3 INSTRUMENTATION

## B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

The RTS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs on other reactor system parameters and equipment performance.

The LSSS, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);

2. Fuel centerline melt shall not occur; and

3. The RCS pressure SL of 2750 psia shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 20 and 10 CFR 100 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR 100

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABLITY

6. Overtemperature ΔT (continued)

as close as possible to 588.4° F. The instrument uncertainty calculations and safety analyses, in combination, have accounted for loop variation in loop specific, full power, indicated ΔT and $T_{avg}$. With respect to $T_{avg}$, a value for T' common to all four loops is permissible within the limits identified in the uncertainty calculations. Outside of those limits, the value of T' will be set appropriately to reflect indicated, loop specific, full power values. In the case of decreasing temperature, the compensated temperature difference shall be no more negative than 3 °F to limit the increase in the setpoint during cooldown transients. The engineering scaling calculations use each of the referenced parameters as an exact gain or reference value. Tolerances are not applied to the individual gain or reference parameters. Tolerances are applied to each calibration module and the overall string calibration. In order to ensure that the Overtemperature ΔT instrument channel is performing in a manner consistent with the assumptions of the safety analyses, it is necessary to verify during the CHANNEL OPERATIONAL TEST that the magnitude of instrument drift from the as-left condition is within limits, and that the input parameters to the trip function are within the appropriate calibration tolerances for the defined calibration conditions (Ref. 9).

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE. Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Overpower $\Delta T$ (continued)

Delta-$T_0$, as used in the overtemperature and overpower $\Delta T$ trips, represents the 100% RTP value as measured for each loop. This normalizes each loop's $\Delta T$ trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in RCS loop $\Delta T$ can be due to several factors, e.g., difference in RCS loop flows and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific $\Delta T$ values. Therefore, loop specific $\Delta T_0$ values are measured as needed to ensure they represent actual core conditions.

The value for T" is a key reference parameter corresponding directly to plant safety analyses initial conditions assumptions for the Overpower $\Delta T$ function. For the purposes of performing a CHANNEL CALIBRATION, the values for $K_4$, $K_5$, $K_6$, and T" are utilized in the safety analyses without explicit tolerances, but should be considered as nominal values for instrument settings. That is, while an exact setting is not expected, a setting as close as reasonably possible is desired. Note that for T", the value for the hottest RCS loop will be set as close as possible to 588.4° F. The instrument uncertainty calculations and safety analyses, in combination, have accounted for loop variation in loop specific, full power, indicated $\Delta T$ and $T_{avg}$. With respect to $T_{avg}$, a value for T" common to all four loops is permissible within the limits identified in the uncertainty calculations. Outside of those limits, the value of T" will be set appropriately to reflect indicated, loop specific, full power values. The engineering scaling calculations use each of the referenced parameters as an exact gain or reference value. Tolerances are not applied to the individual gain or reference parameters. Tolerances are applied to each calibration module and the overall string calibration. In order to ensure that the Overpower $\Delta T$ instrument channel is performing in a manner consistent with the assumptions of the safety analyses, it is necessary to verify during the CHANNEL OPERATIONAL TEST that the magnitude of instrument drift from the as-left condition is within limits, and that the input parameters to the trip function are within the appropriate calibration tolerances for defined calibration conditions (Ref. 9). Note that for the parameter $K_5$, in the case of

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Overpower ΔT (continued)

decreasing temperature, the gain setting must be ≥ 0 to prevent generating setpoint margin on decreasing temperature rates. Similarly, the setting for $K_6$ is required to be equal to 0 for conditions where T ≤ T".

The LCO requires four channels of the Overpower ΔT trip Function to be OPERABLE. Note that the Overpower ΔT trip Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

8. Pressurizer Pressure

The same sensors (PI-0455A, B, & C, PI-0456, PI-0456A, PI-0457, PI-0457A, PI-0458, PI-0458A) provide input to the Pressurizer Pressure — High and — Low trips and the Overtemperature ΔT trip. Since the Pressurizer Pressure channels are also used to provide input to the Pressurizer Pressure Control System, the actuation logic must be able to withstand an input failure to

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Turbine Trip — Low Fluid Oil Pressure (continued)

Below the P-9 setpoint, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, there is no potential for a turbine trip, and the Turbine Trip — Low Fluid Oil Pressure trip Function does not need to be OPERABLE.

b. Turbine Trip — Turbine Stop Valve Closure

The Turbine Trip — Turbine Stop Valve Closure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip from a power level above the P-9 setpoint, approximately 50% power. Below the P-9 setpoint this action will not actuate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. This trip Function will not and is not required to operate in the presence of a single channel failure. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure — High Trip Function, and RCS integrity is ensured by the pressurizer safety valves. This trip Function is diverse to the Turbine Trip — Low Fluid Oil Pressure trip Function. Each turbine stop valve is equipped with one limit switch that inputs to the RTS. If all four limit switches indicate that the stop valves are all closed, a reactor trip is initiated.

The Nominal Trip Setpoint for this Function is set to assure channel trip occurs when the associated stop valve is not fully open (approximately 3.3% closed).

Since the stop valves are designed to fully close once tripped, any indication that the valve is no longer fully open is sufficient to determine the trip status. Because the stop valves close so quickly, any indication near the fully open position (such as 90% open) provides sufficient assurance that the stop valve is going closed. Therefore, for this function, the allowable value was established as an operability limit for the channel operational test.

The LCO requires four Turbine Trip — Turbine Stop Valve Closure channels, one per valve, to be

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1 (continued)

outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the power range channel output every 24 hours. If the calorimetric heat balance results exceed the power range channel output by more than +2% RTP, the power range channel is not declared inoperable, but must be adjusted consistent with the calorimetric heat balance results. If the power range channel output cannot be properly adjusted, the channel is declared inoperable.

If the calorimetric is performed at part power (< 50% RTP), adjusting the power range channel indication in the increasing direction will assure a reactor trip below the safety analysis limit of 118% RTP. Making no adjustment to the power range channel in the decreasing power direction due to a part-power calorimetric assures a reactor trip consistent with the safety analyses.

This allowance does not preclude making indication power adjustments, if desired, when the calorimetric heat balance calculation is less than the power range channel output. To provide close agreement between indicated and calorimetric power and to preserve operating margin, the power range channels are normally adjusted when operating at or near full power during steady-state conditions. However, discretion must be exercised if the power range channel output is adjusted in the decreasing power direction due to a part-power calorimetric (< 50% RTP). This action may introduce a nonconservative bias at higher power levels which may result in an NIS reactor trip above the safety analysis limit of 118% RTP. The cause of the potential nonconservative bias is the decreased accuracy of the calorimetric at reduced power conditions. The primary error

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

contributor to the instrument uncertainty for a secondary side power calorimetric measurement is the feedwater flow measurement which is typically a ΔP measurement across a feedwater venturi. While the measurement uncertainty remains constant in ΔP as power decreases, when translated into flow, the uncertainty increases as a square term. Thus a 1% flow error at 100% RTP can approach a 10% error at 30% RTP even though the ΔP error has not changed. An evaluation of extended operation at part-power conditions would conclude that it is prudent to administratively adjust the setpoint of the Power Range Neutron Flux – High bistables to ≤ 90% RTP for a calorimetric power determined below 50% RTP, and to ≤ 75% RTP for a calorimetric power determined below 20% RTP when: 1) the power range channel output is adjusted in the decreasing power direction due to a part-power calorimetric; or 2) for a post-refueling startup.

Before the Power Range Neutron Flux – High bistables are reset to the nominal value in Table 3.3.1-1 of Specification 3.3.1, the power range channel adjustment must be confirmed based on a calorimetric performed at a power level ≥ 50% RTP.

The Note clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 12 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are inaccurate.

The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate that a difference between the calorimetric heat balance calculation and the power range channel output of more than +2% RTP is not expected in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 EFPD. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. If the NIS channel cannot be properly readjusted, the channel is declared

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3 (continued)

inoperable. This surveillance is primarily performed to verify the (AFD) input to the overtemperature ΔT function.

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 EFPD. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This surveillance is primarily performed to verify the f(AFD) input to the overtemperature ΔT function.

The Note clarifies that the Surveillance is required only if reactor power is ≥ 15% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP.

Axial offset is the difference between the power in the top half of the core and the bottom half of the core expressed as a fraction (percent) of the total power being produced by the core. Mathematically, it is expressed as:

$$AO = 100 \times \frac{(Flux_T - Flux_B)}{(Power)(Flux_T + Flux_B)}$$

where $Flux_T$ = neutron flux at the top of the core, and

$Flux_B$ = neutron flux at the bottom of the core

The relationship between AFD and axial offset is:

$$AFD = AO \times (Power(\%)/100)$$

AFD as displayed on the main control board and as determined by the plant computer use inputs from the power range NIS detectors which are located outside the reactor vessel. Axial offset is measured using incore detectors.

The surveillance assures that the AFD as displayed on the main control board and as determined by the plant computer is within 3% of the AFD as calculated from the axial offset equation. Agreement is required so that the reactor is operated within the bounds of the safety analysis regarding axial power distribution.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3 (continued)

The Frequency of every 31 EFPD is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

SR 3.3.1.4

SR 3.3.1.4 is the performance of a TADOT every 31 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

The RTB test shall include separate verification of the undervoltage and shunt trip mechanisms. Independent verification of RTB undervoltage and shunt trip function is not required for the bypass breakers. No capability is provided for performing such a test at power. The independence test for bypass breakers is included in SR 3.3.1.13. The bypass breaker test shall include a local shunt trip. A Note has been added to indicate that this test must be performed on the bypass breaker prior to placing it in service.

The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

(continued)

THIS PAGE APPLICABLE TO UNIT 2 ONLY.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.5 (continued)

SR 3.3.1.5 is modified by the following Note: The surveillance interval for the Memories Test portion of the ACTUATION LOGIC TEST and the test of the Power Range Block of the Source Range Neutron Flux Trip Block for the Unit 2 Train B SSPS can be extended to the Unit 2 end-of-cycle 10 refueling outage or the next Unit 2 shutdown to MODE 5, whichever comes first.

(continued)

THIS PAGE APPLICABLE TO UNIT 2 ONLY.

BASES (continued)

This page intentionally left blank.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This surveillance is primarily performed to verify the f(AFD) input to the overtemperature ΔT function.

Two Notes modify SR 3.3.1.6. Note 1 states that this Surveillance is required only if reactor power is > 75% RTP and that 7 days is allowed for performing the first surveillance after reaching 75% RTP. Note 2 states that neutron detectors are excluded from the calibration.

The Frequency of 92 EFPD is adequate. It is based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every 92 days.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be within the Allowable Values specified in Table 3.3.1-1.

The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

The "as found" and "as left" values must also be recorded and reviewed for consistency with the assumptions of Reference 7.

This Surveillance Requirement is modified by two Notes that apply only to the Source Range instrument channels. Note 1 requires that the COT include verification that interlocks P-6 and P-10 are in the required state for the existing unit

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.7 (continued)

conditions. Note 2 provides a 4 hour delay in the requirement to perform this surveillance for source range instrumentation when entering Mode 3 from Mode 2. This Note allows a normal shutdown to proceed without delay for the performance of this SR to meet the applicability requirements in Mode 3. This delay allows time to open the RTBs in Mode 3 after which this SR is no longer required to be performed. If the unit is to be in Mode 3 with the RTBs closed for greater than 4 hours, this surveillance must be completed prior to the expiration of the 4 hours.

The Frequency of 92 days is justified in Reference 7.

SR 3.3.1.8

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7, except the frequency is prior to reactor startup. This SR is not required to be met when reactor power is decreased below P-10 (10% RTP) or when MODE 2 is entered from MODE 1 during controlled shutdowns. The Surveillance is modified by a Note that specifies this surveillance can be satisfied by the performance of a COT within 31 days prior to reactor startup. This test ensures that the NIS source range, intermediate range, and power range low setpoint channels are OPERABLE prior to taking the reactor critical.

SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT and is performed every 92 days, as justified in Reference 7.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to RCP undervoltage and underfrequency relays, setpoint verification requires elaborate bench calibration and is accomplished during the CHANNEL CALIBRATION.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.10

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology. The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 18 months is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology for some instrument functions, and the need to perform this Surveillance for some instrument functions under the conditions that apply during a plant outage and the potential for an unplanned plant transient if the Surveillance were performed at power. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable.

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, every 18 months. This SR is modified by a Note that states that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors includes a normalization of the detectors based on a power calorimetric and flux map performed above 75% RTP. The CHANNEL CALIBRATION for the source range neutron detectors includes obtaining the detector preamp discriminator curves and evaluating those curves.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.11 (continued)

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.

SR 3.3.1.12

SR 3.3.1.12 is the performance of a COT of RTS interlocks every 18 months.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.1.13

SR 3.3.1.13 is the performance of a TADOT of the Manual Reactor Trip and the SI Input from ESFAS. This TADOT is as described in SR 3.3.1.4, except that the test is performed every 18 months.

The manual reactor trip TADOT shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the manual reactor trip function. This test shall also verify the OPERABILITY of the Bypass breaker trip circuit(s), including the automatic undervoltage trip.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.13 (continued)

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of the turbine stop valve closure Turbine Trip Functions. This TADOT is as described in SR 3.3.1.4, except that this test is performed after each entry into MODE 3 for a unit shutdown and prior to exceeding the P-9 interlock trip setpoint. Note 1 states that this Surveillance is not required if it has been performed within the previous 31 days. Note 2 states that verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test ensures that the reactor trip on turbine trip Function is OPERABLE prior to entering the Mode of Applicability (above the P-9 power range neutron flux interlock) for this instrument function. The frequency is based on the known reliability of the instrumentation that generates a reactor trip after the turbine trips, and has been shown to be acceptable through operating experience.

SR 3.3.1.15

SR 3.3.1.15 verifies that the individual channel/train actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in FSAR, Chapter 16 (Ref. 8). Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment reaches the required functional state (i.e., control and shutdown rods fully inserted in the reactor core).

For channels that include dynamic transfer Functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer function set to one or with the time constants set to their nominal value. The results must be compared to properly defined acceptance criteria. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.15 (continued)

Response time may be verified by actual response time tests in any series of sequential, overlapping, or total channel measurements; or by the summation of allocation sensor, signal processing, and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable resonse time tests (hydraulic, noise, or power interrupt tests), (2) in place, onsite, or offsite (e.g., vendor) test measurements, or (3) using vendor engineering specifications. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," (Ref. 10), provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

WCAP-14036-P Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," (Ref. 11), provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing assembly of a transmitter.

As appropriate, each channel's response must be verified every 18 months on a STAGGERED TEST BASIS. Testing of the final actuation devices is included in the testing. Response times cannot be determined during unit operation because equipment operation is required to measure response

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.15 (continued)

times. Experience has shown that these components usually pass this surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.3.1.15 is modified by a Note stating that neutron detectors are excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

SR 3.3.1.16

SR 3.3.1.16 is the performance of a COT for the low fluid oil pressure portion of the Turbine Trip Functions as described in SR 3.3.1.7 except that the Frequency is after each entry into MODE 3 for a unit shutdown and prior to exceeding the P-9 interlock trip setpoint. The surveillance is modified by two Notes. Note 1 states that the surveillance may be satisfied if performed within the previous 31 days. Note 2 states that verification of the setpoint is not required. Performance of this test ensures that the reactor trip on turbine trip function is OPERABLE prior to entering the Mode of Applicability (above the P-9 power range neutron flux interlock) for this instrument function. The frequency is based on the known reliability of the instrumentation that generates a reactor trip after the turbine trips, and has been shown to be acceptable through operating experience.

REFERENCES

1. FSAR, Chapter 7.

(continued)

BASES

REFERENCES (continued)

2. FSAR, Chapter 6.

3. FSAR, Chapter 15.

4. IEEE-279-1971.

5. 10 CFR 50.49.

6. WCAP-11269, Westinghouse Setpoint Methodology for Protection Systems; as supplemented by:

   - Amendments 34 (Unit 1) and 14 (Unit 2), RTS Steam Generator Water Level – Low Low, ESFAS Turbine Trip and Feedwater Isolation SG Water Level – High High, and ESFAS AFW SG Water Level – Low Low.

   - Amendments 48 and 49 (Unit 1) and Amendments 27 and 28 (Unit 2), deletion of RTS Power Range Neutron Flux High Negative Rate Trip.

   - Amendments 60 (Unit 1) and 39 (Unit 2), RTS Overtemperature ΔT setpoint revision.

   - Amendments 57 (Unit 1) and 36 (Unit 2), RTS Overtemperature and Overpower ΔT time constants and Overtemperature ΔT setpoint.

   - Amendments 43 and 44 (Unit 1) and 23 and 24 (Unit 2), revised Overtemperature and Overpower ΔT trip setpoints and allowable values.

   - Amendments 104 (Unit 1) and 82 (Unit 2), revised RTS Intermediate Range Neutron Flux, Source Range Neutron Flux, and P-6 trip setpoints and allowable values.

   - Amendments 127 (Unit 1) and 105 (Unit 2), revised Overtemperature ΔT trip setpoint to limit value of the compensated temperature difference and revised the modifier for axial flux difference.

   - Amendments 128 (Unit 1) and 106 (Unit 2), revised Overtemperature ΔT and Overpower ΔT trip setpoints to increase the fundamental setpoints $K_1$ and $K_4$, and to modify coefficients and dynamic compensation terms.

7. WCAP-10271-P-A, Supplement 1, May 1986.

8. FSAR, Chapter 16.

(continued)

BASES

REFERENCES (continued)

9. Westinghouse Letter GP-16696, November 5, 1997.

10. WCAP-13632-P-A Revision 2, "Elimination of Periodic Sensor Response Time Testing Requirements," January 1996.

11. WCAP-14036-P-A Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," October 1998.

12. WCAP-14333-P-A, Rev. 1, October 1998.

13. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.

THIS PAGE APPLICABLE TO UNIT 2 ONLY.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1 (continued)

channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and reliability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and that there is an intact voltage signal path to the master relay coils. The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.2.2 is modified by the following Note: The surveillance interval for the Memories Test portion of the ACTUATION LOGIC TEST and the portions of the ACTUATION LOGIC TEST for Feedwater Isolation on P14 or SI that pass through the memories circuits for the Unit 2 Train B SSPS can be extended to the Unit 2 end-of-cycle 10 refueling outage or the next Unit 2 shutdown to MODE 5, whichever comes first.

(continued)

BASES

LCO (continued)

18. Reactor Vessel Water Level

Reactor Vessel Water Level (LT1310, LT1311, LT1312, LT1320, LT1321, & LT1322) is provided for verification and long term surveillance of core cooling. It is also used for accident diagnosis and to determine reactor coolant inventory adequacy. A RVLIS channel consists of Full Range, Upper Range, and Dynamic Range transmitters. LT1310 and LT1320 are Upper Range, LT1311 and LT1321 are Full Range, and LT1312 and LT1322 are Dynamic Range.

The Reactor Vessel Water Level Monitoring System provides a direct measurement of the collapsed liquid level above the uppercore plate. The collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. Measurement of the collapsed water level is selected because it is a direct indication of the water inventory.

19. Hydrogen Monitors

Hydrogen Monitors (Loops 12979 & 12980) are provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. This variable is also important in verifying the adequacy of mitigating actions.

21. Containment Isolation Valve Position

CIV Position is provided for verification of Containment OPERABILITY, and Phase A isolation.

When used to verify Phase A isolation, the important information is the isolation status of the containment penetrations. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each active containment isolation valve in a containment penetration flow path, i.e., two total channels of containment isolation valve position indication for a penetration flow path with two active valves. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve, as applicable, and prior knowledge of a passive valve, or via system boundary status. If a normally active CIV is known to be closed and deactivated, position

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

The RCS water inventory balance must be performed with the reactor at steady state operating conditions. Therefore, this SR is not required to be performed in MODES 3 and 4 until 12 hours of steady state operation have been established. In all cases, this SR is required to be performed prior to entering MODE 2 to ensure the assessment of RCS leakage prior to critical operation.

Steady state operation is required to perform a proper inventory balance; calculations during maneuvering are not useful and a Note requires the Surveillance to be performed when steady state is established. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The 12 hour Frequency after steady state operation has been achieved provides for those situations where a transient occurs, and the duration of the transient is such that the 72 hour Frequency plus the 25% extension allowed by SR 3.0.2 would be exceeded. In this event, the SR would be due within 12 hours after steady state operation has been reestablished.

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

(continued)

BASES (continued)

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. The accumulators will discharge following a large main steam line break, however, their impact is minor with respect to this limiting design basis event. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 24 hours. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 24 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions. The 24 hours allowed to restore an inoperable accumulator to OPERABLE status is justified in WCAP-15049-A, Rev. 1 (Ref. 5)

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and pressurizer pressure reduced to

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. Sampling the affected accumulator within 6 hours after a 1% volume increase (7% of indicated level) will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), because the water contained in the RWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 6).

SR 3.5.1.5

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 two accumulators 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.

(continued)

BASES (continued)

REFERENCES

1. Deleted.
2. FSAR, Chapter 6.
3. 10 CFR 50.46.
4. FSAR, Chapter 15.
5. WCAP-15049-A, Rev. 1, April 1999.
6. NUREG-1366, February 1990.