Attachment 1

Surry Power Station

Proposed Technical Specification Changes

U. Unrestricted Area

An unrestricted area shall be any area at or beyond the site boundary where access is not controlled by the licensee for purpose of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

V. Member(s) of the Public

Member(s) of the public shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category shall not include non-employees such as vending machine servicemen or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

W. Core Operating Limits Report

The Core Operating Limits Report is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.2.C. Plant operation within these operating limits is addressed in individual specifications.

conservative, than the loci of points of THERMAL POWER, coolant system average temperature, and coolant system pressure for which either the calculated DNBR is equal to the design DNBR limit or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the calculated DNBR reaches the design DNBR limit and, thus, this arbitrary limit is conservative with respect to maintaining clad integrity. The plant conditions required to violate these limits are precluded by the protection system and the self-actuated safety valves on the steam generator. Upper limits of 70% power for loop stop valves open and 75% with loop stop valves closed are shown to completely bound the area where clad integrity is assured. These latter limits are arbitrary but cannot be reached due to the Permissive 8 protection system setpoint which will trip the reactor on high nuclear flux when only two reactor coolant pumps are in service.

Operation with natural circulation or with only one loop in service is not allowed since the plant is not designed for continuous operation with less than two loops in service.

TS Figure 2.1-1 is based on an $F_{\Delta H}(N)$ of 1.62, a 1.55 cosine axial flux shape, and a deterministic DNB analysis procedure including margin to accommodate rod bowing[1]. TS Figure 2.1-1 is also bounding for a statistical treatment of key DNBR analysis parameter uncertainties including an enthalpy rise hot channel factor which follows the following functional form: $F_{\Delta H}(N) = 1.56\ [1 + 0.3(1\text{-}P)]$ where P is the fraction of rated power. TS Figures 2.1-2 and 2.1-3 are based on an $F_{\Delta H}(N)$ of 1.55, a deterministic treatment of key DNB analysis parameter uncertainties, and include a 0.2 rather than 0.3 part power multiplier for the enthalpy rise hot channel factor. The $F_{\Delta H}(N)$ limit presented in the unit- and reload-specific CORE OPERATING LIMITS REPORT is confirmed for each reload to be accommodated by the Reactor Core Safety Limits.

These hot channel factors are higher than those calculated at full power over the range between that of all control rod assemblies

E. Minimum Temperature for Criticality

Specifications

1. Except during LOW POWER PHYSICS TESTS, the reactor shall not be made critical at any Reactor Coolant System temperature above which the moderator temperature coefficient is more positive than the limit specified in the CORE OPERATING LIMITS REPORT.

2. In no case shall the reactor be made critical with the Reactor Coolant System temperature below DTT + 10°F, where the value of DTT + 10°F is as determined in Part B of this specification.

3. When the Reactor Coolant System temperature is below the minimum temperature as specified in E-1 above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to primary coolant depressurization.

4. The reactor shall not be made critical when the Reactor Coolant System temperature is below 522°F.

Basis

During the early part of a fuel cycle, the moderator temperature coefficient may be calculated to be slightly positive at coolant temperatures in the power operating range. The moderator coefficient will be most positive at the beginning of cycle life, when the boron concentration in the coolant is the greatest. Later in the cycle, the boron concentration in the coolant will be lower and the moderator coefficient will be less positive or will be negative in the power operating range. At the beginning of cycle life, during pre-operational physics tests, measurements are made to determine that the moderator coefficient is less than the limit specified in the CORE OPERATING LIMITS REPORT.

The requirement that the reactor is not to be made critical when the moderator coefficient is greater than the low power limit specified in the CORE OPERATING LIMITS REPORT has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant pressure. This requirement is waived during LOW POWER PHYSICS TESTS to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operation precautions will be taken. In addition, the strong negative Doppler coefficient [2][3] and the small integrated Delta k/k would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical with a Reactor Coolant System temperature below DTT + 10°F provides increased assurance that the proper relationship between Reactor Coolant System pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil ductility transition temperature range. Heatup to this temperature is accomplished by operating the reactor coolant pumps.

The requirement that the reactor is not to be made critical with a Reactor Coolant System temperature below 522°F provides added assurance that the assumptions made in the safety analyses remain bounding by maintaining the moderator temperature within the range of those analyses.

If a specified shutdown reactivity margin is maintained (TS Section 3.12), there is no possibility of an accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure.

(1) UFSAR Figure 3.3-8
(2) UFSAR Table 3.3-1
(3) UFSAR Figure 3.3-9

## 3.12 CONTROL ROD ASSEMBLIES AND POWER DISTRIBUTION LIMITS

### Applicability

Applies to the operation of the control rod assemblies and power distribution limits.

### Objective

To ensure core subcriticality after a reactor trip, a limit on potential reactivity insertions from hypothetical control rod assembly ejection, and an acceptable core power distribution during power operation.

### Specification

A. Control Bank Insertion Limits

1. Whenever the reactor is critical, except for physics tests and control rod assembly exercises, the shutdown control rods shall be fully withdrawn.

2. Whenever the reactor is critical, except for physics tests and control rod assembly exercises, the full length control rod banks shall be inserted no further than the appropriate limit specified in the CORE OPERATING LIMITS REPORT.

3. The Control Bank Insertion Limits shown in the CORE OPERATING LIMITS REPORT may be revised on the basis of physics calculations and physics data obtained during unit startup and subsequent operation in accordance with the following:

    a. The sequence of withdrawal of the controlling banks, when going from zero to 100% power, is A, B, C, D.
    b. An overlap of control banks, consistent with physics cal-

B. Power Distribution Limits

1. At all times except during LOW POWER PHYSICS TESTS, the hot channel factors defined in the basis meet the following limits:

$F_Q(Z) \le (CFQ/P) \times K(Z)$ for $P > 0.5$
$F_Q(Z) \le (CFQ/0.5) \times K(Z)$ for $P \le 0.5$

where: CFQ = the FQ limit at RATED POWER specified in the CORE OPERATING LIMITS REPORT,

$$P = \frac{\text{THERMAL POWER}}{\text{RATED POWER}}$$, and

K(Z) = the normalized FQ limit as a function of core height, Z, as specified in the CORE OPERATING LIMITS REPORT

$F\Delta H(N) \le CFDH \times (1 + PFDH \times (1-P))$

where: CFDH = the FΔH(N) limit at RATED POWER specified in the CORE OPERATING LIMITS REPORT,

PFDH = the Power Factor Multiplier for FΔH(N) specified in the CORE OPERATING LIMITS REPORT, and

$$P = \frac{\text{THERMAL POWER}}{\text{RATED POWER}}$$

2. Prior to exceeding 75% power following each core loading and during each effective full power month of operation thereafter, power distribution maps using the movable detector system shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this confirmation:

   a. The measurement of total peaking factor $F_Q^{Meas}$ shall be increased by eight percent to account for manufacturing tolerances, measurement error and the effects of rod bow. The measurement of enthalpy rise hot channel factor $F_{\Delta H}^N$ shall be compared directly to the limit specified in Specification 3.12.B.1. If any measured hot channel factor exceeds its limit specified under Specification 3.12.B.1, the reactor power and high neutron flux trip setpoint shall be reduced until the limits under Specification 3.12.B.1 are met. If the hot channel factors cannot be brought to within the FQ(Z) and $F_{\Delta H}^N$ limits as specified in the CORE OPERATING LIMITS REPORT within 24 hours, the Overpower ΔT and Overtemperature ΔT trip setpoints shall be similarly reduced.

   b. The provisions of Specification 4.0.4 are not applicable.

Amendment Nos.

It should be noted that the enthalpy rise factors are based on integrals and are used as such in the DNB and LOCA calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in radial (x-y) power shapes throughout the core. Thus, the radial power shape at the point of maximum heat flux is not necessarily directly related to the enthalpy rise factors. The results of the loss of coolant accident analyses are conservative with respect to the ECCS acceptance criteria as specified in 10 CFR 50.46 using the upper bound $F_Q(Z)$ times the hot channel factor normalized operating envelope given in the CORE OPERATING LIMITS REPORT.

When an $F_Q$ measurement is taken, measurement error, manufacturing tolerances, and the effects of rod bow must be allowed for. Five percent is the appropriate allowances for measurement error for a full core map (greater than or equal to 38 thimbles, including a minimum of 2 thimbles per core quadrant, monitored) taken with the movable incore detector flux mapping system, three percent is the appropriate allowance for manufacturing tolerances, and five percent is appropriate allowance for rod bow. These uncertainties are statistically combined and result in a net increase of 1.08 that is applied to the measured value of $F_Q$.

In the $F^N_{\Delta H}$ limit specified in the CORE OPERATING LIMITS REPORT, there is a four percent error allowance, which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq$ CFDH [1 + PFDH (1-P)]/1.04. The 4% allowance is based on the considerations that (a) normal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting $F_Q$, (b) the operator has a direct influence on $F_Q$ through movement of rods and can limit it to the desired value; he has no direct control over $F^N_{\Delta H}$, and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests and which may influence $F_Q$, can be compensated for by tighter axial control. An appropriate allowance for the measurement uncertainty for $F^N_{\Delta H}$ obtained from a full core map (≥ 38 thimbles, including a minimum of 2 detectors per core quadrant, monitored) taken with the movable incore detector flux mapping system has been incorporated in the statistical DNBR limit. Measurement of the hot channel factors are required as part of startup physics tests, during each effective full power power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following core loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would, otherwise, affect these bases.

For normal operation, it has been determined that, provided certain conditions are observed, the enthalpy rise hot channel factor $F^N_{\Delta H}$ limit will be met. These conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position. An indicated misalignment limit of 13 steps precludes a rod misalignment no greater than 15 inches with consideration of maximum instrumentation error.

2. Control rod banks are sequenced with overlapping banks as shown in the Control Bank Insertion Limits specified in the CORE OPERATING LIMITS REPORT.

3. The full length Control Bank Insertion Limits specified in the CORE OPERATING LIMITS REPORT are not violated.

4. Axial power distribution control procedures, which are given in terms of flux difference control and Control Bank Insertion Limits are observed. Flux differences refers to the difference

b. The moderator temperature coefficient in the power operating range is less than or equal to the limits specified in the CORE OPERATING LIMITS REPORT.

c. Capable of being made subcritical in accordance with Specification 3.12.A.3.C.

B. Reactor Coolant System

1. The design of the Reactor Coolant System complies with the code requirements specified in Section 4 of the UFSAR.

2. All piping, components, and supporting structures of the Reactor Coolant System are designed to Class 1 seismic requirements, and have been designed to withstand:

   a. Primary operating stresses combined with the Operational seismic stresses resulting from a horizontal ground acceleration of 0.07g and a simultaneous vertical ground acceleration of 2/3 the horizontal, with the stresses maintained within code allowable working stresses.

   b. Primary operating stresses when combined with the Design Basis Earthquake seismic stresses resulting from a horizontal ground acceleration of 0.15g and a simultaneous vertical ground

## 6.2 GENERAL NOTIFICATION AND REPORTING REQUIREMENTS

### Specification

A. The following actions shall be taken for Reportable Events:

1. A report shall be submitted pursuant to the requirements of Section 50.73 to 10 CFR, and

2. Each Reportable Event shall be reviewed by the SNSOC. The Vice President - Nuclear Operations and the MSRC shall be notified of the results of this review.

B. Immediate notifications shall be made in accordance with Section 50.72 to 10 CFR.

C. CORE OPERATING LIMITS REPORT

Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle. Parameter limits for the following Technical Specifications are defined in the CORE OPERATING LIMITS REPORT:

1. TS 3.1.E and TS 5.3.A.6.b - Moderator Temperature Coefficient

2. TS 3.12.A.2 and TS 3.12.A.3 - Control Bank Insertion Limits

3. TS 3.12.B.1 and TS 3.12.B.2 - Power Distribution Limits

The analytical methods used to determine the core operating limits identified above shall be those previously reviewed and approved by the NRC, and identified below. The core operating limits shall be determined so that applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided for information for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

REFERENCES

1. VEP-FRD-42, Rev. 1-A, "Reload Nuclear Design Methodology," September 1986

   (Methodology for TS 3.1.E and TS 5.3.A.6.b - Moderator Temperature Coefficient; TS 3.12.A.2 and 3.12.A.3 - Control Bank Insertion Limit; TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor and Nuclear Enthalpy Rise Hot Channel Factor)

2a. WCAP-9220-P-A, Rev. 1, "Westinghouse ECCS Evaluation Model - 1981 Version," February 1982 (W Proprietary)

   (Methodology for TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor)

2b. WCAP-9561-P-A, ADD. 3, Rev. 1, "BART A-1: A Computer Code for the Best Estimate Analysis of Reflood Transients-Special Report: Thimble Modeling in W ECCS Evaluation Model," July 1986 (W Proprietary)

   (Methodology for TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor)

2c. WCAP-10266-P-A, Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," March 1987 (W Proprietary)

   (Methodology for TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor)

2d. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985 (W Proprietary)

(Methodology for TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor)

2e. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985 (W Proprietary)

(Methodology for TS 3.12.B.1 and TS 3.12.B.2 - Heat Flux Hot Channel Factor)

3a. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," June 1987

(Methodology for TS 3.12.B.1 and TS 3.12.B.2 - Nuclear Enthalpy Rise Hot Channel Factor)

3b. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code," July 1990

(Methodology for TS 3.12.B.1 and TS 3.12.B.2 - Nuclear Enthalpy Rise Hot Channel Factor)

Attachment 2

Surry Power Station

Discussion and Significant Hazards Consideration

## Discussion of Changes

### Introduction

In October of 1988, the NRC issued Generic Letter 88-16[1] to provide guidance for the implementation of a Core Operating Limits Report (COLR). The COLR facilitates modification of certain Technical Specification core operating parameter limits via the provisions of 10 CFR 50.59, provided the methodologies utilized to establish the revised limits have been reviewed and approved by the NRC and are listed in the Technical Specifications. The ability to redefine core operating limits without a formal change to the Technical Specifications will provide increased flexibility for optimization of reload core designs, and will reduce the engineering and regulatory burden associated with implementation of plant operating parameter changes made with NRC-approved methodologies.

### Background

Technical Specification changes to incorporate a COLR into the North Anna Units 1 and 2 Technical Specifications have been previously submitted[2] and approved[3] by the NRC.

A safety evaluation and proposed Technical Specification changes have been prepared to support implementation of a COLR at Surry Units 1 and 2. The proposed Technical Specification changes were developed in accordance with the guidelines presented in NRC Generic Letter 88-16[1] and are discussed in the following section.

According to the guidance provided by NRC Generic Letter 88-16[1], implementation of a COLR requires:

1. the addition of a Technical Specification definition for a formal report (COLR) which includes a list of COLR parameters and the Technical Specification Limiting Condition for Operation (LCO) in which the parameter is specified,

2. the addition of a Technical Specification administrative reporting requirement to submit the formal COLR containing the cycle-specific parameter limits to the NRC for informational purposes, and

3. the modification of individual Technical Specification LCOs to indicate that cycle-specific paramaters shall be maintained within the limits provided in the COLR.

The methodologies utilized to establish the modified limits must have been reviewed and approved by the NRC, and must be listed in the Technical Specifications.

The proposed Technical Specification changes described in the following section address the Generic Letter 88-16[1] requirements described above, and are consistent with the requirements of 10 CFR 50.36, "Technical Specifications."

## Technical Specification Changes

### General

The Technical Specification changes described herein apply to Surry Units 1 and 2. Core operating limit parameters affected by the proposed changes include:

1. Moderator Temperature Coefficient (TS 3.1.E and TS 5.3.A.6.b)

2. Control Bank Insertion Limits (TS 3.12.A.2 and TS 3.12.A.3)

3. Power Distribution Limits: FQ(Z), FΔH(N), and part power multiplier (TS 3.12.B.1 and TS 3.12.B.2)

Specific Technical Specification changes have been prepared in accordance with the guidance of NRC Generic Letter 88-16[1].

### COLR Definition - TS 1.0.W

TS 1.0.W has been added to the Technical Specifications to define the COLR.

### Clarification of Basis Statement for Safety Limits and Limiting Safety System Settings - TS 2.1 (Basis)

A statement has been added to the Basis section for TS 2.1 to clarify the relationship of the Reactor Core Safety Limits of TS Figures 2.1-1, 2.1-2, and 2.1-3 to the cycle-specific radial peaking factor (FΔH) limit presented in the COLR. Specifically, it is noted that the FΔH(N) limit presented in the unit- and reload-specific CORE OPERATING LIMITS REPORT is confirmed for each reload to be accommodated by the Reactor Core Safety Limits.

TS Figures 2.1-2 and 2.1-3 provide Reactor Core Safety Limits for two-loop operation. Although the statement of clarification described above is applicable to these two figures, two-loop operation is presently prohibited by TS 3.3.A.11.

Moderator Temperature Coefficient - TS 3.1.E and Basis

This Specification has been revised to state that the cycle-specific value of the Moderator Temperature Coefficient limit shall be maintained within the limits identified in the COLR.

Control Bank Insertion Limits - TS 3.12.A.2, TS 3.12.A.3, and TS Figures 3.12-1A, 3.12-1B, 3.12-4A, and 3.12-4B

These Specifications have been revised and their associated Technical Specification figures deleted to reflect the inclusion of cycle-specific Control Bank Insertion Limits in the COLR.

Power Distribution Limits - TS 3.12.B.1, TS 3.12.B.2, TS Figure 3.12-8, and Bases

These Specifications and their bases have been revised, and their associated Technical Specification figures deleted to reflect the inclusion of cycle-specific hot channel factor ($F_Q(Z)$ and $F_{\Delta H}(N)$) limits in the COLR.

Moderator Temperature Coefficient (Design Features; Reactor) - TS 5.3.A.6.b

This Technical Specification Design Feature has been revised to state that the cycle-specific value of the Moderator Temperature Coefficient limit shall be maintained within the limits identified in the COLR.

Administrative Reporting Requirement - TS 6.2.C

TS 6.2.C has been added to the Technical Specifications to define a new administrative NRC reporting requirement for the COLR. The reporting requirement is consistent with that presented in NRC Generic Letter 88-16[1].

Administrative Changes

Defined words have been capitalized. System names have been corrected for consistency. Grammatical errors and misspellings have been corrected throughout the Technical Specifications affected by the proposed changes.

## Safety Significance

10 CFR 50.36 provides regulatory guidance for the development of Technical Specifications which define parameter limits and minimum functional requirements for plant equipment. The Technical Specification requirements reflect the parameters and system performance characteristics which have been demonstrated via safety analysis to ensure that safety analysis acceptance criteria are met. The methodologies used to calculate and evaluate these parameters have been reviewed and approved by the NRC.

Under the proposed Technical Specifications, parameter limits for certain reload-dependent parameters will be specified in the COLR. The NRC-approved methodologies listed in the proposed Technical Specifications will be used to calculate and evaluate the parameter limits presented in the COLR for each reload core.

The removal of parameter limits for reload-dependent parameters from the Technical Specifications has no impact upon plant operation or safety. No safety-related equipment, safety function, or plant operating characteristic will be altered as a result of this proposed change. Because (a) the applicable UFSAR limits will be maintained, (b) the Technical Specifications will continue to require operation within the core operational limits calculated by NRC-approved methodologies, and (c) appropriate actions to be taken if limits are violated will remain in the Technical Specifications, it may be concluded that the proposed changes are administrative in nature and that the Technical Specifications remain in conformance with 10 CFR 50.36.

Following incorporation of the COLR into the Technical Specifications, any changes to COLR parameter values will be made in accordance with the provisions of 10 CFR 50.59. No changes to the Technical Specifications will be required.

<u>Summary</u>

Technical Specification changes to implement a COLR at Surry have been made in accordance with regulatory guidance presented in Generic Letter 88-16[1]. The Technical Specification changes associated with the removal of parameter limits for certain reload-dependent parameters from the Technical Specifications, and inclusion of these parameters in the COLR are administrative in nature. The Surry Technical Specifications remain in conformance with the requirements of 10 CFR 50.36 following the incorporation of the proposed Technical Specification changes.

COLRs for Surry 1 Cycle 12 (S1C12) and Surry 2 Cycle 12 (S2C12) are presented in Appendices A and B, respectively.

## References


(1) "Removal of Cycle-Specific Parameter Limits from Technical Specifications," USNRC Generic Letter 88-16, dated October 4, 1988.

(2) Letter from W. L. Stewart (Virginia Power) to USNRC, "Virginia Electric and Power Company; North Anna Power Station Units 1 and 2; Implementation of Generic Letter 88-16," Serial No. 90-030, dated March 29, 1990.

(3) Letter from L. B. Engle (USNRC) to W. L. Stewart, "North Anna Units 1 and 2 - Issuance of Amendments Re: Core Operating Limits Report TAC Nos. 76828 and 76829), dated June 7, 1991.

APPENDIX A

CORE OPERATING LIMITS REPORT (COLR)
FOR SURRY 1 CYCLE 12