UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 189
License No. DPR-32

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated July 2, 1993, as supplemented December 10, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-32 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 189 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 2, 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 189
License No. DPR-37

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated July 2, 1993, as supplemented December 10, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-37 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 189 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 2, 1994

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 189 TO FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 189 TO FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

| Remove Pages | Insert Pages |
|---|---|
| 1.0-7 | 1.0-7 |
| 2.1-4 | 2.1-4 |
| 3.1-18 | 3.1-18 |
| 3.1-19 | 3.1-19 |
| 3.12-1 | 3.12-1 |
| 3.12-2 | 3.12-2 |
| 3.12-3 | 3.12-3 |
| 3.12-4 | 3.12-4 |
| 3.12-16 | 3.12-16 |
| 3.12-17 | 3.12-17 |
| 3.12-18 | 3.12-18 |
| 5.3-3 | 5.3-3 |
| 6.2-1 | 6.2-1 |
| --- | 6.2-2 |
| --- | 6.2-3 |

S. SITE BOUNDARY

The site boundary shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee.

T. 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 qauarters or for industrial, commerical, institutional, or recreational purposes.

U. 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.

V. 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 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: $F\Delta H(N) = 1.56\ [1 + 0.3(1-P)]$ 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. The maximum upper limit for the moderator temperature coefficient shall be:

   a. + 6 pcm/°F at less than 50% of RATED POWER, or
   b. + 6 pcm/°F at 50% of RATED POWER and linearly decreasing to 0 pcm/°F at RATED POWER.

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 surveillance testing, the shutdown control rod assemblies shall be fully withdrawn. With a shutdown control rod assembly not fully withdrawn, within 1 hour either fully withdraw the assembly or declare the assembly inoperable and apply Specification 3.12.C.

2. Whenever the reactor is critical, except for physics tests and control rod assembly surveillance testing, the full length control banks shall be inserted no further than the appropriate limit specified in the CORE OPERATING LIMITS REPORT. With a control bank inserted beyond the limits shown in the applicable figure, restore the control rod assembly bank to within its limits within 2 hours, or reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED POWER which is allowed by the group position using TS Figures 3.12.1A or 1B, or place the reactor in HOT SHUTDOWN within 6 hours.

3. The Control Bank Insertions 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 control banks, when going from zero to 100% power, is A, B, C, D.

b. An overlap of control banks, consistent with physics calculations and physics data obtained during unit startup and subsequent operation, will be permitted.

c. The shutdown margin with allowance for a stuck control rod assembly shall be greater than or equal to 1.77% reactivity under all steady-state operation conditions, except for physics tests, from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at HOT SHUTDOWN ($T_{avg} \geq 547°F$) if all control rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no changes in xenon or boron.

4. Whenever the reactor is subcritical, except for physics tests, the critical control rod assembly position, i.e., the control rod assembly position at which criticality would be achieved if the control rod assemblies were withdrawn in normal sequence with no other reactivity changes, shall not be lower than the insertion limit for zero power.

5. Insertion limits do not apply during physics tests or during periodic surveillance testing of control rod assemblies. However, the shutdown margin indicated above must be maintained except for the LOW POWER PHYSICS TEST to measure control and shutdown bank worth and shutdown margin. For this test the reactor may be critical with all but one full length control rod assembly, expected to have the highest worth, inserted.

6. With a maximum of one control or shutdown bank inserted beyond the insertion limit specified in Specification 3.12.A.2 during control rod assembly testing pursuant to Specification 4.1, and immovable due to a failure of the Rod Control System, POWER OPERATION

may continue* provided that:

a. the affected bank insertion is limited to 18 steps below the insertion limit as measured by the group step counter demand position indicators,

b. the affected bank is trippable,

c. each control rod assembly is aligned to within ± 12 steps of its respective group step counter demand position indicator,

d. The shutdown margin requirement of Specification 3.12.A.3.c is determined to be met at least every 12 hours thereafter, and

e. the affected bank is restored to within the insertion limits of Specification 3.12.A within 72 hours.

Otherwise place the unit in HOT SHUTDOWN within the next 6 hours.

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,

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

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

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

* Provision for continued operation does not apply to Control Bank D inserted beyond the insertion limit.

2. Prior to exceeding 75% of RATED 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 $\Delta T$ and Overtemperature $\Delta T$ trip setpoints shall be similarly reduced within the next 4 hours.

   b. The provisions of Specification 4.0.4 are not applicable.

3. The reference equilibrium indicated axial flux difference (called the target flux difference) at a given power level $P_O$ is that indicated axial flux difference with the core in equilibrium xenon conditions (small or no oscillation) and the control rod assemblies more than 190 steps withdrawn. The target flux difference at any other power level P is equal to the target value at $P_O$ multiplied by the ratio $P/P_O$. The target flux difference shall be measured at least once per equivalent full power quarter. The target flux difference must be updated during each effective full power month of operation either by actual measurements or by linear interpolation using the most recent value and the value predicted for the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

4. Except as modified by Specifications 3.12.B.4.a, b, c, or d below, the indicated axial flux difference shall be maintained within a ± 5% band about the target flux difference (defines the target band on axial flux difference).

In addition to the above, the peak linear power density and the nuclear enthalpy rise hot channel factor must not exceed their limiting values which result from the large break loss of coolant accident analysis based on the Emergency Core Cooling System acceptance criteria limit of 2200°F on peak clad temperature. This is required to meet the initial conditions assumed for the loss of coolant accident. To aid in specifying the limits of power distribution, the following hot channel factors are defined:

> $F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerance on fuel pellets and rods.
>
> $F_Q^E$, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod, and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux for non-statistical applications.
>
> $F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power for both loss of coolant accident and non-loss of coolant accident considerations.

It should be noted that the enthalpy rise factors are based on integrals and are used as such in the DNB and loss of coolant accident 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 Emergency Core Cooling System 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 allowance 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 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 rod assemblies in a single bank move together with no individual control rod assembly insertion differing by more than 15 inches from the bank demand position. An indicated misalignment limit of 13 steps precludes a control rod assembly 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 difference refers to the difference between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and the bottom halves of the core.

The permitted relaxation in $F^N_{\Delta H}$ with decreasing power level allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, this hot channel factor limit is met.

A recent evaluation of DNB test data obtained from experiments of fuel rod bowing in thimble cells has identified that the reduction in DNBR due to rod bowing in thimble cells is more than completely accommodated by existing thermal margins in the core design. Therefore, it is not necessary to continue to apply a rod bow penalty to $F^N_{\Delta H}$.

The procedures for axial power distribution control are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of flux difference ($\Delta I$) and a reference value which corresponds to the full power equilibrium value of axial offset (axial offset = $\Delta I$/fractional power). The reference value of flux difference varies with power level and burnup, but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control given in Specification 3.12.B.4 together with the surveillance requirements given in Specification 3.12.B.2 assure that the Limiting Condition for Operation for the heat flux hot channel factor is met.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with the full length rod control bank more than 190 steps withdrawn (i.e., normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full power at which the core

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. The maximum upper limit for the moderator temperature coefficient shall be:

1) + 6 pcm/°F at less than 50% of RATED POWER, or

2) + 6 pcm/°F at 50% of RATED POWER and linearly decreasing to 0 pcm/°F at RATED POWER.

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

B. <u>Reactor Coolant System</u>

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.

<u>REFERENCES</u>

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)