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June 1, 1992

Posted
Amdt. 169 to DPR-37

Docket Nos. 50-280
and 50-281

Mr. W. L. Stewart
Senior Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

Dear Mr. Stewart:

SUBJECT: SURRY UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: F DELTA H LIMIT AND STATISTICAL DNBR METHODOLOGY (TAC NOS. M81271 AND M82168)

The Commission has issued the enclosed Amendment No. 170 to Facility Operating License No. DPR-32 and Amendment No. 169 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application transmitted by letter dated July 8, 1991, as supplemented April 15, 1992.

These amendments increase the $F_{\Delta h}$ surveillance limit from a value of 1.55 to 1.56 and provide changes for implementation of the statistical DNBR evaluation methodology.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
(Original Signed By)
Bart C. Buckley, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 170 to DPR-32
2. Amendment No. 169 to DPR-37
3. Safety Evaluation

cc w/enclosures:
See next page
*See previous concurrence

OFC	:LA:PDII-2	:PM:PDII-2	:D:PDII-2	:* SRXB	: OGC	:
NAME	:D. Miller	:B. Buckley	:H. Berkow	: R. Jones	:	:
DATE	:6/1/92	: 5/4/92	: 5/8/92	: 04/30/92	: 5/15/92	:

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 170
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 8, 1991, as supplemented April 15, 1992, 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. 170 , 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: June 1, 1992

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 169
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 8, 1991, as supplemented April 15, 1992, 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. 169 , 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: June 1, 1992

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
TS 2.1-4	TS 2.1-4
TS 2.1-5	TS 2.1-5
TS 2.1-6	TS 2.1-6
TS Figure 2.1-1	TS Figure 2.1-1
TS 3.12-3	TS 3.12-3
TS 3.12-11	TS 3.12-11
--	TS 3.12-11a
TS 3.12-14	TS 3.12-14
TS 3.12-15	TS 3.12-15
TS 3.12-16	TS 3.12-16
TS 3.12-19	TS 3.12-19
TS Figure 3.12-8	TS Figure 3.12-8
TS 4.1-9d	TS 4.1-9d

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.

These hot channel factors are higher than those calculated at full power over the range between that of all control rod assemblies fully withdrawn to maximum allowable control rod assembly insertion. The control rod assembly insertion limits are covered by Specification 3.12. Adverse power distribution factors could occur at lower power levels because additional control rod assemblies are in the core; however, the control rod assembly insertion limits dictated by TS Figures. 3.12-1A (Unit 1) and 3.12-1B (Unit 2) ensure that the DNBR is always greater at partial power than at full power.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure and thermal power level that would result in a DNBR less than the design DNBR limit(3) based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 574.4°F and a steady state nominal operating pressure of 2235 psig. For deterministic DNBR analysis, allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions. The steady state nominal operating parameters and allowances for steady state errors given above are also applicable for two loop operation except that the steady state nominal operating power level is less than or equal to 60%.

For statistical DNBR analyses, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability that the minimum DNBR for the limiting rod is greater than or equal to the statistical DNBR limit. The uncertainties in the plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a statistical DNBR limit which must be met in plant safety analyses using values of input parameters without uncertainties. The statistical DNBR limit also ensures that at least 99.9% of the core avoids the onset of DNB when the limiting rod is at the DNBR limit.

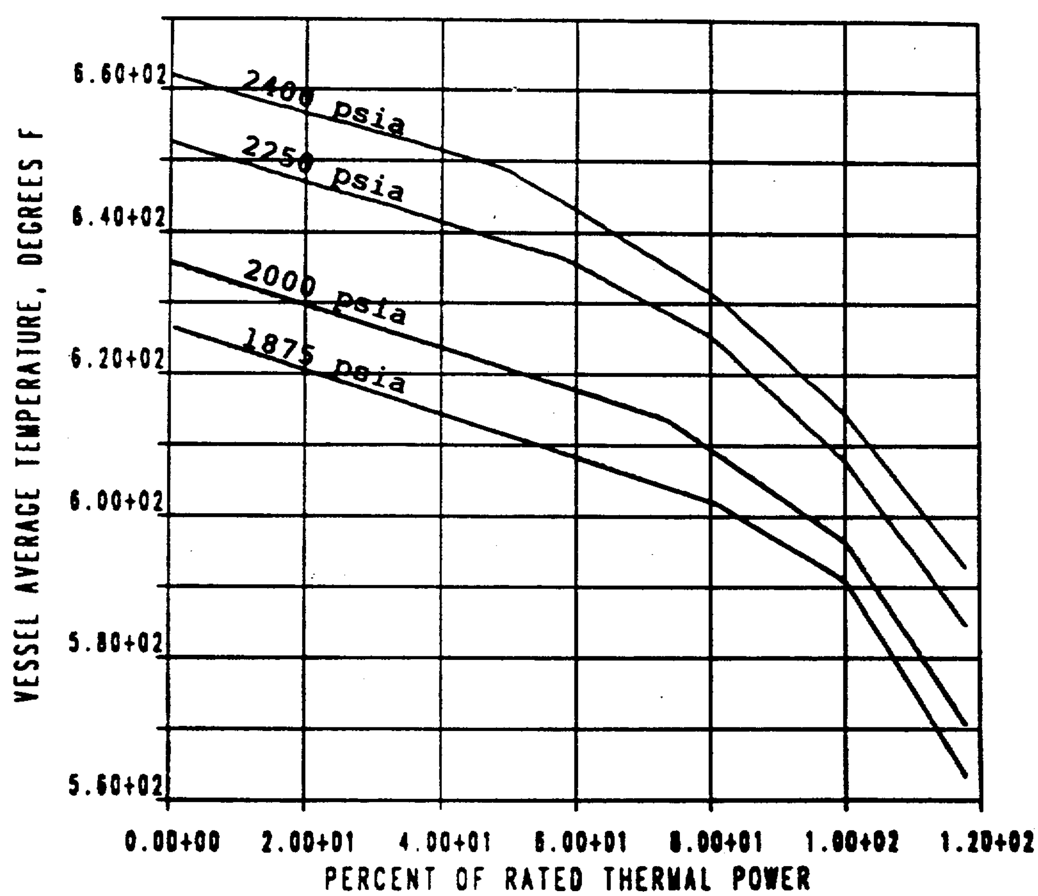
The fuel overpower design limit is 118% of rated power. The overpower limit criterion is that core power be prevented from reaching a value at which fuel pellet melting would occur. The value of 118% power allows substantial margin to this limiting criterion. Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length have been included in the calculation of this limit.

References

1) FSAR Section 3.4
2) FSAR Section 3.3
3) FSAR Section 14.2

TS FIGURE 2.1-1

REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS - THREE LOOP OPERATION, 100% FLOW



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:

$F_Q(Z) \leq 2.32/P \times K(Z)$ for $P > 0.5$

$F_Q(Z) \leq 4.64 \times K(Z)$ for $P \leq 0.5$

$F^N_{\Delta H} \leq 1.56\ [1 + 0.3\ (1\text{-}P)]$ for three loop operation

$\leq 1.55\ [1 + 0.2\ (1\text{-}P)]$ for two loop operation

where P is the fraction of rated power at which the core is operating, K(Z) is the function given in TS Figure 3.12-8, and Z is the core height location of F_Q.

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 $F_{\Delta H}$ 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 limits of $F_Q(Z) \leq 2.32/P \times K(Z)$ and $F^N_{\Delta H} \leq 1.56$ within 24 hours, the Overpower ΔT and Overtemperature ΔT trip setpoints shall be similarly reduced.

c. In hot, intermediate and cold shutdown conditions, the step demand counters shall be operable and capable of determining the group demand positions to within ±2 steps. The rod position indicators shall be available to verify rod movement upon demand.

2. If a rod position indicator channel is out of service, then:

a. For operation above 50% of rated power, the position of the RCC shall be checked indirectly using the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod exceeding 24 steps, or

b. Reduce power to less than 50% of rated power within 8 hours. During operations below 50% of rated power, no special monitoring is required.

3. If more than one rod position (RPI) indicator channel per group or two RPI channels per bank are inoperable during control bank motion to achieve criticality or power operations, then the requirements of Specification 3.0.1 will be followed.

F. <u>DNB PARAMETERS</u>

1. The following DNB related parameters shall be maintained within their limits during power operation:

Reactor Coolant System $T_{avg} \leq 578.4°F$
Pressurizer Pressure ≥ 2205 psig
Reactor Coolant System Total Flow Rate ≥ 273,000 gpm

a. The Reactor Coolant System T_{avg} and Pressurizer Pressure shall be verified to be within their limits at least once every 12 hours.

b. The Reactor Coolant System Total Flow Rate shall be determined to be within its limit by measurement at least once per refueling cycle.

2. When any of the parameters in Specification 3.12.F.1 has been determined to exceed its limit, either restore the parameter to within its limit within 2 hours or reduce reactor power to less than 5% of rated thermal power within the next 4 hours.

3. The limit for Pressurizer Pressure in Specification 3.12.F.1 is not applicable during either a thermal power ramp increase in excess of 5% of rated thermal power per minute or a thermal power step increase in excess of 10% of rated thermal power.

Basis

The reactivity control concept assumed for operation is that reactivity changes accompanying changes in reactor power are compensated by control rod assembly motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated for by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of power is by the control groups. A reactor trip occurring during power operation will place the reactor into the hot shutdown condition. The control rod assembly insertion limits provide for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod assembly remains fully withdrawn, with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit

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 ECCS 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 LOCA and non-LOCA considerations.

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 by TS Figure 3.12-8.

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 specified limit of $F^N_{\Delta H}$, there is a four percent error allowance, which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.56\ [1 + 0.3\ (1\text{-}P)]/1.04$. 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 influent F_Q, can be compensated for by tighter axial control. An appropriate allowance for the measurement uncertainty for $F_{\Delta H}^{N}$ 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 $F_{\Delta H}^{N}$ 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 TS Figures 3.12-1A, 3.12-1B.

3. The full length control bank insertion limits 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

In some instances of rapid unit power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band; however, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for the allowable flux difference at 90% power, in the range ± 13.8 percent (± 10.8 percent indicated) where for every 2 percent below rated power, the permissible flux difference boundary is extended by 1 percent.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished, by using the boron system to position the full length control rods to produce the required indicated flux difference.

A 2% quadrant tilt allows that a 5% tilt might actually be present in the core because of insensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rod and an error allowance. No increase in F_Q occurs with tilts up to 5% because misaligned control rods producing such tilts do not extend to the unrodded plane, where the maximum F_Q occurs.

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the UFSAR assumptions and have been analytically demonstrated to be adequate to maintain a minimum DNBR which is greater than the design limit throughout each analyzed transient. Measurement uncertainties are accounted for in the DNB design margin. Therefore, measurement values are compared directly to the surveillance limits without applying instrument uncertainty.

The 12 hour periodic surveillance of temperature and pressure through instrument readout is sufficient to ensure that these parameters are restored to within their limits following load changes and other expected transient operation. The measurement of the RCS total flow rate once per refueling cycle is adequate to detect flow degradation.

TS FIGURE 3.12-8

HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE

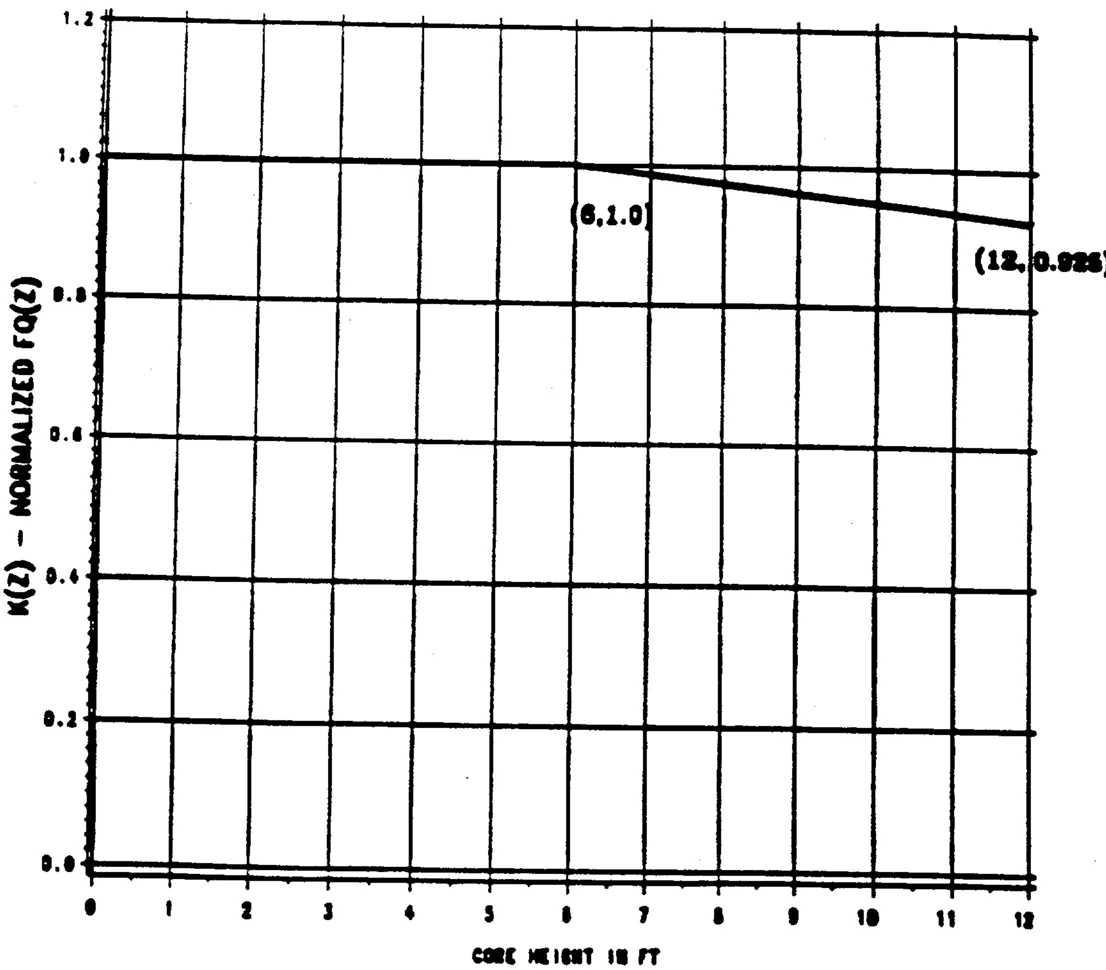


TABLE 4.1-2A (CONTINUED)

MINIMUM FREQUENCY FOR EQUIPMENT TESTS

	DESCRIPTION	TEST	FREQUENCY	FSAR SECTION REFERENCE
18.	Primary Coolant System	Functional	1. Periodic leakage testing [a] on each valve listed in Specification 3.1.C.7a shall be accomplished prior to entering power operation condition after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in cold shutdown condition for 72 hours if testing has not been accomplished in the preceeding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.	
19.	Containment Purge MOV Leakage	Functional	Semi-Annual (Unit at power or shutdown) if purge valves are operated during interval [c]	
20.	Containment Hydrogen Analyzers	a. Channel Check b. Channel Functional Test c. Channel Calibration using sample gas containing: 1. One volume percent (± 0.25%) hydrogen, balance nitrogen 2. Four volume percent (± 0.25%) hydrogen, balance nitrogen 3. Channel calibration test will include startup and operation of the Heat Tracing System	Once per 12 hours Once per 31 days Once per 92 days on staggered basis	
21.	RCS Flow	Flow ≥ 273,000 gpm	Once per refueling cycle	14

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(b) Minimum differential test pressure shall not be below 150 psid.

(c) Refer to Section 4.4 for acceptance criteria.

*See Specification 4.1.D

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

In a submittal dated July 8, 1991, as supplemented April 15, 1992, the Virginia Electric and Power Company (VEPCO or licensee) proposed a change in the enthalpy rise hot channel factor (F-delta-h) for its Surry Units 1 and 2 plants from 1.55 to 1.62. The submittal described the application of the VEPCO statistical departure from nucleate boiling (DNB) methodology to the Surry Units 1 and 2 plants, discussed the impact of the 1.62 F-delta-h value on Surry non-LOCA event analyses, and provided a Surry small break loss-of-coolant accident (LOCA) analysis assuming the new F-delta-h value. The licensee also proposed Technical Specifications (TS) changes to reflect the methods and values discussed in the submittal. The April 15, 1992 letter provided supplemental information that did not change the initial proposed no significant hazards consideration determination.

The proposed F-delta-h increase would accommodate increased radial power factors resulting from installation of flux suppression inserts in Surry Unit 1. These inserts are designed to reduce peripheral core power and thereby reduce reactor vessel neutron radiation embrittlement.

2.0 EVALUATION

2.1 Methodologies

2.1.1 DNB Methodologies

In assessing the impact of the proposed 1.62 F-delta-h value, the licensee referenced the deterministic W-3 DNB methodology (and the deterministic application of the WRB-1 correlation for certain events within its range of applicability) currently applied to the Surry units, and the VEPCO statistical DNB methodology using the WRB-1 DNB correlation (the licensee's July 8, 1991 submittal contains a table identifying events for which this methodology will be used).

With either DNB methodology, the licensee determined a retained margin, the difference between the DNB ratio (DNBR) limit for the methodology and a design limit against which the plant has been explicitly analyzed. The licensee expresses this margin as a percent of the design limit for the methodology and assesses certain DNBR penalties (e.g., rod bow) against it when necessary.

The current W-3 deterministic methodology is applicable to both Westinghouse LOPAR fuel and Surry Improved Fuel (SIF), which are contained in the Surry cores. For application of this methodology to Surry, the licensee has determined a correlation DNBR limit of 1.24 as applicable, and has set a design limit of 1.46. The retained margin using the W-3 deterministic methodology is 18 percent. In cases where the WRB-1 correlation is used deterministically, the DNBR limit is 1.17 and the retained margin is 20 percent.

The statistical DNB methodology used for the Surry F-delta-h determination is described in the topical report VEP-NE-2-A. This methodology is applied only to SIF fuel and was previously approved for application to the VEPCO North Anna plants. The North Anna design is like the Surry design in all aspects pertinent to the applicability of the methodology. The staff therefore finds the statistical DNBR methodology described in VEP-NE-2-A is applicable to Surry Units 1 and 2.

In the application of the VEPCO statistical DNB methodology to Surry, the licensee determined the statistical DNBR limit (SDL) of the correlation using Surry-specific parameters (e.g., for vessel average temperature, pressurizer pressure, thermal power, vessel mass flow) uncertainties in the calculation of statepoint uncertainties. The licensee determined an SDL of 1.27 as applicable to the Surry units, and has set a design limit of 1.46 for consistency with the W-3 deterministic methodology design limit. The retained margin using the VEPCO statistical DNB methodology is 13 percent.

The above methodologies have previously been approved for existing Surry analyses and/or have been approved for application to the North Anna plants which are of similar design. The staff, therefore, finds them applicable to the Surry plants, as described in the licensee's July 8, 1991 submittal.

2.1.2 Small Break LOCA Methodology

The emergency core cooling system (ECCS) small break LOCA evaluation model (EM) with the Westinghouse NOTRUMP code used for the Surry small break reanalysis is described in WCAP-10079-P-A and WCAP-10054-P-A. This approved EM is applicable to the Surry plants.

2.2 F-Delta-H

In its July 8, 1991 submittal, the licensee proposed a design F-delta-h limit of 1.62. The proposed TS surveillance F-delta-h limit is 1.56, considering a 4 percent measurement uncertainty. Evaluation analyses, except those using the VEPCO statistical DNB methodology, assume a 1.62 value. Analyses using the statistical methodology assume the 1.56 value, because the measurement uncertainty is factored into the method.

Using the W-3 deterministic DNB methodology the licensee determined that the increase in F-delta-h to 1.62 would result in a 7.3 percent DNBR penalty. In an assessment of the reactor protection setpoints using the approved methodologies, the licensee determined that the existing TS core thermal limits (CTLs) were not bounding. The licensee constructed new CTLs reflecting the higher F-delta-h, which are presented in proposed TS Figure 2.1-1. Existing overpressure-delta-T (OPDT) and overtemperature-delta-T (OTDT) trip setpoints were found to be adequate. No change in these reactor protection setpoints are proposed.

Because the licensee used acceptable methodologies in making these assessments, the staff finds the resultant determinations regarding reactor protection setpoints acceptable.

2.3 Transient and Accident Analyses

In its July 8, 1991, the licensee provided an assessment of the impact of the proposed F-delta-h change on the Surry Final Safety Analysis Report (FSAR) Chapter 14 design basis event analyses.

2.3.1 Non-LOCA Events

The licensee addressed the impact of the proposed F-delta-h change on non-LOCA events covering both LOPAR fuel and SIF fuel.

For LOPAR fuel, analyzed using the W-3 deterministic methodology, the licensee indicated that existing analyses and protection setpoints bound or include an assumed 1.62 F-delta-h. The most limiting OTDT DNB event was identified to be a rod withdrawal at power with existing OTDT trip setpoints indicated to be adequate to bound the 1.62 F-delta-h assumption. The most limiting DNB event, which does not trip on OTDT, was identified to be a loss of flow event, whose existing analysis assumes a 1.62 F-delta-h.

For SIF fuel, the most limiting OTDT DNB event was identified to be a rod withdrawal at power, for which the licensee indicated that the current analysis is bounding for an assumed 1.62 F-delta-h.

The most limiting DNB event for SIF fuel which does not trip on OTDT was identified to be a loss of flow event. The licensee indicated that the existing analysis using the WRB-1 correlation deterministically is based on a 1.62 F-delta-h value. However, the licensee provided a reanalysis of this event using the VEPCO statistical methodology to enhance the analysis margin and to demonstrate application of the methodology. The calculated minimum DNBR for this event was about 1.5, which is higher than the 1.46 design limit and does not involve retained margin compensation.

The most limiting DNB event for SIF fuel analyzed using deterministic DNB methods was identified to be a locked rotor event. The licensee indicated that it had performed a thermal-hydraulic reanalysis of this event assuming 1.62 F-delta-h for both fuels and concluded that the existing 5 percent failed fuel assumption remains limiting.

The remainder of the non-LOCA Chapter 14 events are discussed in the licensee's July 8, 1991 submittal and indicate that DNBRs are not significantly reduced by the 1.62 F-delta-h, not affected by the change in F-delta-h, or not applicable to the present Surry core.

2.3.2 LOCA Analyses

The licensee indicates that the current large break LOCA of record assumes a 1.62 F-delta-h. The calculated peak cladding temperature (PCT) in that analysis is 1979°F.

The July 8, 1991 submittal provides the results of a small break LOCA reanalysis using the Westinghouse NOTRUMP code and assuming a 1.65 F-delta-h value. The calculated PCT was 1504°F. This is much lower than the large break PCT. Small break LOCAs continue to be less limiting than large break LOCAs with the 1.62 (or 1.65) F-delta-h assumptions.

2.3.3 Analysis Conclusions

Based on the assessments provided by the licensee, the staff concludes that Surry operation will continue to be bounded by Chapter 14 analyses with the F-delta-h raised to 1.62.

3.0 TS Changes

The licensee's submittal proposed the following TS changes to reflect the 1.62 F-delta-h value and the methodologies used to assess its impact.

a. TS 2.1-4, - change in discussion of TS Figure 2.1-1 to reflect 1.62 F-delta-h and statistical methodology implementation.

b. TS 2.1-5 - change in discussion of DNBR analyses to reflect differences in use of statistical DNB methodology versus deterministic DNBR methodology.

c. TS 2.1-6 - continuation of changes from previous page.

d. TS Figure 2.1-1 - change to reactor core thermal and hydraulic safety limits to reflect 1.62 F-delta-h.

e. TS 3.12-3 - change to equation for F(N)-delta-h to reflect F-delta-h surveillance limit; change in line referring to above equation; change to F-delta-h surveillance limit value in discussion of maintenance operation within hot channel factor limits.

f. TS 3.12-11 - adds surveillance requirements for DNB-related parameters: reactor coolant system (RCS) average temperature, pressurizer pressure, and RCS total flow rate, to reflect use of statistical DNBR methodology.

g. TS 3.12-11a - continuation of changes from previous page.

h. TS 3.12-14 - adds qualification to discussion of engineering heat flux hot channel factor (FQE) to clarify that the FQE penalty is applicable only in non-statistical analyses, to reflect use of statistical methodology.

i. TS 3.12-15 - Bases discussion of F-delta-h is updated to reflect 1.56 surveillance limit and use of statistical methodology.

j. TS 3.12-16 - continuation of changes from previous page.

k. TS 3.12-19 - Bases discussion is added for DNB parameters specified on TS pages 3.12-11 and 3.12-11a.

l. TS Figure 3.12-8 - change to hot channel factor normalized operating envelope, to reflect changed F-delta-h.

m. TS Table 4.1-2A - adds RCS flow to table of minimum frequency for equipment tests, to reflect use of statistical methodology.

These TS changes reflect use of the methodologies discussed in Section 2.1.1 and an increased F-delta-h value. The staff finds the TS changes acceptable because they are consistent with similar changes implemented at the North Anna plants, which are of like design.

4.0 SUMMARY

As discussed in Section 2.1.1, the staff finds that the W-3 and WRB-1 deterministic DNB methodologies are applicable to the Surry units, as limited in their present use, based on their currently approved usage. The staff finds that the VEPCO statistical DNB methodology is applicable to the Surry units based on its currently approved applicability to the North Anna plants of similar design.

Based on the justifications provided by the licensee, the staff finds the 1.62 enthalpy rise hot channel factor (F-delta-h) acceptable for operation of the Surry units with LOPAR and SIF fuels.

The staff also finds the proposed TS changes which accommodate the methodological and operational changes acceptable.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendments. The State official had no comment.

7.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding (56 FR 47246). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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