June 30, 1989

Docket Nos. 50-338
and 50-339

DISTRIBUTION
See attached sheet

Mr. W. R. Cartwright
Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

Dear Mr. Cartwright:

SUBJECT: NORTH ANNA UNITS 1 AND 2 (NA-1&2) - APPROVAL OF CONTINUED USE OF NEGATIVE MODERATOR COEFFICIENT FOR NA-1 AND ISSUANCE OF AMENDMENT FOR NA-2 (TAC NOS. 71071 AND 71072)

The Commission has issued the enclosed Amendment No. 100 to Facility Operating License No. NPF-7 for the North Anna Power Station, Unit No. 2 (NA-2). The amendment revises the Technical Specifications (TS) in response to your letter dated June 17, 1987.

This amendment revises the NA-2 Technical Specifications (TS) in accordance with Virginia Electric and Power Company's Statistical DNBR Evaluation Methodology for a less restrictive negative moderator temperature coefficient.

Also, the continued use of these TS at NA-1 for Cycle 8 and Cycle 9, etc., is hereby approved.

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

Leon B. Engle, Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 100 to NPF-7
2. Safety Evaluation

cc w/enclosures:
See next page

[NA-1&2 AMEND 71071/71072]

LA:PDII-2	PM:PDII-2	D:PDII-2	OGC
DMiller	LEngle:bd	HBerkow	
06/21/89	06/27/89	06/ /89	06/29/89

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Mr. W. R. Cartwright
Virginia Electric & Power Company

North Anna Power Station
Units 1 and 2

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

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 100
License No. NPF-7

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

   A. The application for amendment by Virginia Electric and Power Company, et al., (the licensee) dated June 17, 1987, 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 2.C.(2) of Facility Operating License No. NPF-7 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 100 , 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 the 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 30, 1989

TO FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Page

B2-1
3/4 1-5
3/4 1-6
3/4 3-10
B3/4 1-2
B3/4 2-5
B3/4 2-6

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through a correlation. The DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, 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 DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. As an additional criterion, meeting the DNBR limit also ensures that at least 99.9% of the core avoids the onset of DNB when the plant is at the DNBR limit.

The curves of Figures 2.1-1, 2.1-2, and 2.1-3 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the design limit DNBR, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.49 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.49\ [1+0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f(delta I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, were initially designed to ANSI B 31.1 1967 Edition and ANSI B 31.7 1969 Edition (Table 5.2.1-1 of FSAR) which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. For the all rods withdrawn, beginning of core life condition $<0.6 \times 10^{-4}$ $\Delta k/k/°F$ below 70 percent RATED THERMAL POWER $\leq 0.0 \times 10^{-4}$ $\Delta k/k/°F$ at or above 70 percent RATED THERMAL POWER.

b. Less negative than -5.0×10^{-4} $\Delta k/k/°F$ for all the rods withdrawn, end of core life at RATED THERMAL POWER.

APPLICABILITY: Specification 3.1.1.4.a - MODES 1 and 2* only#
Specification 3.1.1.4.b - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the limit of 3.1.1.4.a above, operations in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to within its limit within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. The control rods are maintained within the withdrawal limits established above until subsequent measurement verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of 3.1.1.4.b above, be in HOT SHUTDOWN within 12 hours.

*With $K_{eff} \geq 1.0$
#See Special Test Exception 3.10.3

MODERATOR TEMPERATURE COEFFICIENT

SURVEILLANCE REQUIREMENTS

4.1.1.4 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL Limit of Specification 3.1.1.4.a above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. The MTC shall be measured at any THERMAL POWER and compared to -4.0×10^{-4} delta k/k/°F (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicated the MTC is more negative than -4.0×10^{-4} delta k/k/°F, the MTC shall be remeasured, and compared to the EOL MTC limit of specification 3.1.1.4.b, at least once per 14 EFPD during the remainder of the fuel cycle.[1]

(1) Once the equilibrium boron concentration (all rods withdrawn, RATED THERMAL POWER condition) is 60 ppm or less, further measurement of the MTC in accordance with 4.1.1.4.b may be suspended providing that the measured MTC at an equilibrium boron concentration of ≤ 60 ppm is less negative than -4.7×10^{-4} $\Delta k/k/°F$.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION	SETPOINT	ALLOWABLE VALUES	FUNCTION
P-7 (Cont'd)	3 of 4 Power range below setpoint and	8%	>7%	Prevents reactor trip on: Low flow or reactor coolant pump breakers open in more than one loop, Undervoltage (RCP busses), Underfrequency (RCP busses), Turbine Trip, Pressurizer low pressure, and Pressurizer high level.
	2 of 2 Turbine Impulse chamber pressure below setpoint (Power level decreasing)	8%	>7%	
P-8	2 of 4 Power range above setpoint (Power level increasing)	30%	<31%	Permit reactor trip on low flow or reactor coolant pump breaker open in a single loop.
	3 of 4 Power range below setpoint (Power level decreasing)	28%	>27%	Blocks reactor trip on low flow or reactor coolant pump breaker open in a single loop.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	$\leq$ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	$\leq$ 0.5 seconds*
7. Overtemperature ΔT	$\leq$ 4.0 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 2.0 seconds
11. Pressurizer Water Level--High	$\leq$ 2.0 seconds

*Neutron detectors are exempt from response time testing. Response of the neutron flux signal portion of the channel time shall be measured from detector output or input of first electronic component in channel.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.77% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM, as provided by either one RCP or one RHR pump as required by Specification 3.4.1.1, provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 9957 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control. The requirement that certain valves remain closed at all times except during planned boron dilution or makeup, activities provides assurance that an inadvertent boron dilution will not occur.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed for this parameter in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value was obtained by incrementally correcting the MTC used in the FSAR analyses to nominal operating conditions. These corrections involved adding the incremental change in the MTC associated with a core condition of Bank D inserted to an all rods withdrawn condition and an incremental change in MTC to account for measurement uncertainty at RATED THERMAL POWER conditions. These corrections result in the limiting MTC value of -5.0 x 10^{-4} delta k/k/°F. The MTC value of -4.0 x 10^{-4} delta k/k/°F represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -5.0 x 10^{-4} delta k/k/°F.

Once the equilibrium boron concentration falls below about 60 ppm, dilution operations take an extended amount of time and reliable MTC measurements become more difficult to obtain due to the potential for fluctuating core conditions over the test interval. For this reason, MTC measurements may be suspended provided the measured MTC value at an equilibrium full power boron concentration $\leq$60 ppm is less negative than -4.7 x 10^{-4} delta k/k/°F. The difference between this value and the limiting MTC value of -5.0 x 10^{-4} delta k/k/°F conservatively bounds the maximum credible change in MTC between the 60 ppm equilibrium boron concentration (all rods withdrawn, RATED THERMAL POWER conditions) and the licensed end of cycle, including the effect of boron concentration, burnup, and end-of-cycle coastdown.

The surveillance requirements for measurement of the MTC at the beginning and near the end of each fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, and 3) the P-12 interlock is above its setpoint, 4) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 5) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

When $F^N_{\Delta H}$ is measured, 4% is the appropriate experimental error allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ contains a 4% error allowance. Normal operation will result in a measured $F^N_{\Delta H}$ less than or equal to 1.49. The 4% allowance is based on the following considerations:

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q,

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

Fuel rod bowing reduces the value of the DNB ratio. Credit is available to offset this reduction in the margin available between the safety analysis design DNBR value (1.46 for Virginia Electric and Power Company statistical methods) and the limiting design DNBR value (1.26 for Virginia Electric and Power Company statistical methods). A discussion of the rod bow penalty is presented in the FSAR.

The hot channel factor $F_Q M(Z)$ is measured periodically and increased by a cycle and height dependent power factor, N(Z), to provide assurance that the limit on the hot channel factor, $F_Q(Z)$, is met. N(Z) accounts for the non-equilibrium effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The N(Z) function for normal operation is provided in the Core Surveillance Report per Specification 6.9.1.7.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of 4 symmetric thimbles. The two sets of 4 symmetric thimbles is a unique set of 8 detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

3/4.2.5 DNB PARAMETERS

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 initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR greater than the design limit throughout each analyzed transient. Measurement uncertainties must be accounted for in the DNB design margin.

The 12 hour periodic surveillance of these parameters thru instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 100 TO

FACILITY OPERATING LICENSE NOS. NPF-4 AND NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-338 AND 50-339

INTRODUCTION

By letter dated June 17, 1987, the Virginia Electric and Power Company (the licensee) submitted proposed changes to the Technical Specifications (TS) for the North Anna Power Station, Units No. 1 and 2 (NA-1&2). The changes support the implementation of the licensee's Statistical DNBR (Departure from Nucleate Boiling Ratio) Evaluation Methodology as documented in Topical Report VEP-NE-2 of the same name. The principal Updated Final Safety Analysis Report (UFSAR) Chapter 15 DNB events have been analyzed under the new DNB methodology. Some TS changes are needed as a result of these analyses.

Several NA-1&2 TS need to be changed to incorporate the revised DNBR ratio limit and the results of the associated transient analysis. The proposed changes to the NA-1&2 TS include a less restrictive negative moderator temperature coefficient (MTC) limit.

The new methodology employs Monte Carlo methods to evaluate the DNBR sensitivity to key parameters. The methodology is similar to those which have been approved by NRC for some of the vendors and uses the WRB-1 CHF correlation. The topical report "VEP-NE-2" was approved by the NRC on May 28, 1987. We found the methodology acceptable provided that four conditions were met when a plant-specific submittal was made. These four conditions are:

1. The choice of "Nominal Statepoints" must be justified.

2. For the statistically treated parameters, the uncertainty distributions must be justified.

3. The model uncertainty must be substantiated.

4. Topical Report COBRA/WRB-1 (VEP-NE-3) must be approved.

2.0 DISCUSSION

In December 1988, the MTC TS change was required for continued operation of NA-1 for the remainder of Cycle 7. At that time our review of the licensee's submittal had proceeded to the point that we could approve the MTC TS change for the remainder of NA-1 Cycle 7. On January 3, 1989, Amendment No. 112 was issued to Facility Operating License No. NPF-7 for NA-1 which approved the MTC TS for the remainder of Cycle 7. We have now completed our review for NA-2. Each of the four conditions stated in the NRC Safety Evaluation (SE) for VEP-NE-2 is addressed below.

2.1 Nominal Statepoints

To satisfy the condition on nominal statepoints, the choice of the nominal statepoints must be shown to maximize the DNBR standard deviation (and therefore, the DNBR limit) over the proposed range of applicability. The licensee's approach was to perform Monte Carlo calculations at core thermal limit and low flow statepoints. These conditions spanned the pressure range between the high and low trip setpoints, inlet temperature between a boundary cooldown event and a maximum heatup, power up to the 118 percent overpower limit and a boundary low flow event. The analysis consisted of 10 sets of 2000 calculations, each performed over the full range of normal operation and anticipated transient conditions. The standard deviations were then plotted as a function of statepoint temperature. The data showed a clear dependence on temperature. A regression analysis was then performed and the residuals were plotted showing no trends. This indicated that the standard deviation was a function of temperature only. Therefore, the limiting statepoint was specified to be Statepoint 5 with power equal to 118 percent, inlet temperature equal to 538.6°F, pressurizer pressure equal to 1860 psia, and flow equal to 92 percent. The DNBR mean was 1.28 with a standard deviation of 0.1572. This satisfies the condition on selection of nominal statepoints.

2.2 Uncertainty of Statistically Treated Parameters

The statistically treated parameters are core power, pressurizer pressure, inlet temperature, vessel mass flow, core bypass flow and the nuclear and engineering enthalpy rise factors. The uncertainties for core power, pressurizer pressure, inlet temperature and vessel flow are quantified in WCAP-1203. The analysis was performed under the standard Westinghouse uncertainty analysis methodology.

The total core bypass flow and its uncertainty was confirmed as being bounded by the NA-1&2 core uprating Improved Thermal Design Procedure (ITDP) analysis values. Because of the difficulty in characterizing the form of the uncertainty distribution, the implementation analysis assumed that the probability was uniformly spread over a much larger range than was justified by the sum of the components.

The nuclear enthalpy-rise factor uncertainty was based on available measurement/predictive data, which consisted of over 11,000 points taken from 9 cycles of operation at both NA-1&2. The error of prediction relative to the radial power factor measurement ([P-M]/M) yielded a mean value of 0.1 percent with a standard deviation of 1.55 percent. The non-zero mean is conservatively positive and as a conservative measure, a standard deviation of 2 percent was used in the implementation analysis. The data was shown to be a normal distribution curve.

The engineering enthalpy rise uncertainty factor consists primarily of the uncertainty in hot channel power and flow. These factors were quantified by means of a closed-channel calculation, in which boundary values of high hot channel power and low flow were employed. A uniformly distributed 2 percent uncertainty was found to conservatively bound the results. This satisfies the condition on the uncertainty of statistically treated parameters.

2.3 Model Uncertainty

The model uncertainty was included to account for differences between the 6 channel COBRA model, which was used for the Monte Carlo calculations, and the 25 channel COBRA production model, used for performing DNBR calculations. Comparisons show that the 6 channel model DNBR standard deviation is consistently much larger than the standard deviation produced by the 25 channel production model. However, a model uncertainty was quantified as an upper confidence limit on the S (model) where the S (model) is the standard deviation on the ratio of the 6 to 25 channel model DNBR using 100 random statepoints. This satisfies the condition on model uncertainty.

2.4 COBRA/WRB-1 Verification (VEP-NE-3)

Our review of VEP-NE-3 has been completed. Our SER approving the topical COBRA/WRB-1 (VEP-NE-3) was dated June 14, 1989.

2.5 TS Changes

As noted above, implementation of the licensee's DNBR evaluation methodology requires that the NA-1&2 TS be updated to reflect the change in the plant DNBR licensing basis. These TS changes are described below.

2.5.1 TS 3/4.1.1 Most Negative MTC Limit

The change in the Limiting Condition for Operation (LCO 3.1.1.4) and the Surveillance Requirements (SR 4.1.1.4) for the MTC are intended to provide an end-of-cycle and associated trigger values which are appropriate for current NA-1&2 fuel cycles.

The revised limit and trigger values are based on a revised safety analysis of the UFSAR Chapter 15 transients which are sensitive to the most negative MTC parameter. It is noted that the DNB design limit was not violated in any of the analyzed transients.

2.5.2 TS 3/4.3.1 Pressurizer Water Level Response Time

The change to Table 3.3-2, which is referenced by LCO 3.3.1.1, reflects a new requirement to have the pressurizer water level response time be less than or equal to 2.0 seconds. A response time requirement has been added to protect against filling the pressurizer prior to the actuation of the overtemperature delta T reactor trip.

2.5.3 TS B 2.1.1 Full Core DNB Probability Criterion

The change in bases Section 2.1.1 adds a new criterion in the establishment of the DNBR limit for the licensee's Statistical DNB Methodology. Previously, traditional analyses and ITDP analyses considered only peak pin DNB probability. Plant operation required that, for normal operation and Condition II operation, the peak pin avoid DNB with 95 percent probability at a 95 percent confidence level. The new methodology retains this criterion, and adds an additional criterion that the DNB probability of every rod, when summed over the whole core, shows that at least 99.9 percent of the core is expected to remain in the nucleate boiling regime.

2.5.4 TS B 3/4.1 MTC Surveillance Requirements

The changes to bases Section 3/4.1 are related to the new MTC limit and surveillance requirements. Additionally, the reference to the moderator density coefficient (MDC) is deleted because it is no longer related to the safety analyses performed by the licensee. The MDC parameter was used in the previous safety analyses performed by the licensee's fuel vendor, Westinghouse Electric Corporation. Since the safety analyses performed by the licensee uses temperature instead of density to specify moderator reactivity feedback, it is preferable to use MTC in the TS.

An additional benefit of this approach is that the relationship between the TS limit and the safety analysis limit can be more clearly defined. In fact, the two limits differ only by the measurement uncertainty and a correction for Bank D insertion. The revised bases section makes the connection between the two limits clearer.

2.5.5 TS B 3/4.2.3 DNBR Limits

TS B 3/4.2.3 has been modified to reflect the revised DNBR limit as obtained with the new methodology. The new safety analysis DNBR limit is 1.26; the addition of 13.7 percent retained DNBR margin yields a design DNBR limit of 1.46. Separate values were not derived for the typical and thimble cell, since the single limit was shown to be bounding for both. The retained margin is used for such applications as compensation of the rod bow penalty, for example.

The rod bow penalty is Westinghouse proprietary information and is not listed in the FSAR. However, a thorough discussion of the rod bow penalty is provided in the FSAR, and the sources of the appropriate numerical values are referenced.

2.5.6 TS B 3/4.2.5 DNB Parameter Surveillance

The change to bases Section 3/4.2.5 clarifies the treatment of measurement uncertainties. The licensee has performed analyses to show that the measurement uncertainties on the DNB parameters can be offset by the retained DNB margin and need not be accounted for by the plant operations staff.

3.0 EVALUATION

Based on our review of the licensee's June 17, 1987 submittal, we conclude that the requested TS changes are acceptable. This conclusion is based on the fact that the plant-specific implementation submittal for use of the licensee's Statistical DNBR Methodology adequately addresses three of the four conditions specified in the NRC Safety Evaluation dated May 28, 1987 for the staff's review of VEP-NE-2, "Statistical DNBR Evaluation Methodology." In addition, the staff's review of the fourth condition, the approval of "COBRA/WRB-1 (VEP-NE-3)," is now complete and the topical has been approved. Therefore, based on all of the above, the TS changes described above are acceptable for NA-2. In addition, these TS changes, as noted above, were previously approved for NA-1 for Cycle 7 only and continued use of these TS is hereby approved for NA-1 Cycle 8, Cycle 9, etc.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to surveillance requirements. The staff has determined that the amendment involves 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 published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets 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 the amendment.

CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: June 30, 1989

Principal Contributor:

Margaret Chatterton

DATED: June 30, 1989

AMENDMENT NO. 100 TO FACILITY OPERATING LICENSE NO. NPF-7-NORTH ANNA UNIT 2

Docket File
NRC & Local PDRs
PDII-2 Reading
S. Varga, 14/E/4
G. Lainas, 14/H/3
H. Berkow
D. Miller
L. Engle
OGC
D. Hagan, 3302 MNBB
E. Jordan, 3302 MNBB
B. Grimes, 9/A/2
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