ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATIONS CHANGES

NORTH ANNA UNIT 1

VIRGINIA ELECTRIC AND POWER COMPANY

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1* for 3 loop operation and 2.1-2 and 2.1-3 for 2 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

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* For the period of operation until steam generator replacement, the combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1a.

Nominal T_{avg} = 586.8°F

Nominal RCS flow = 268,500 GPM

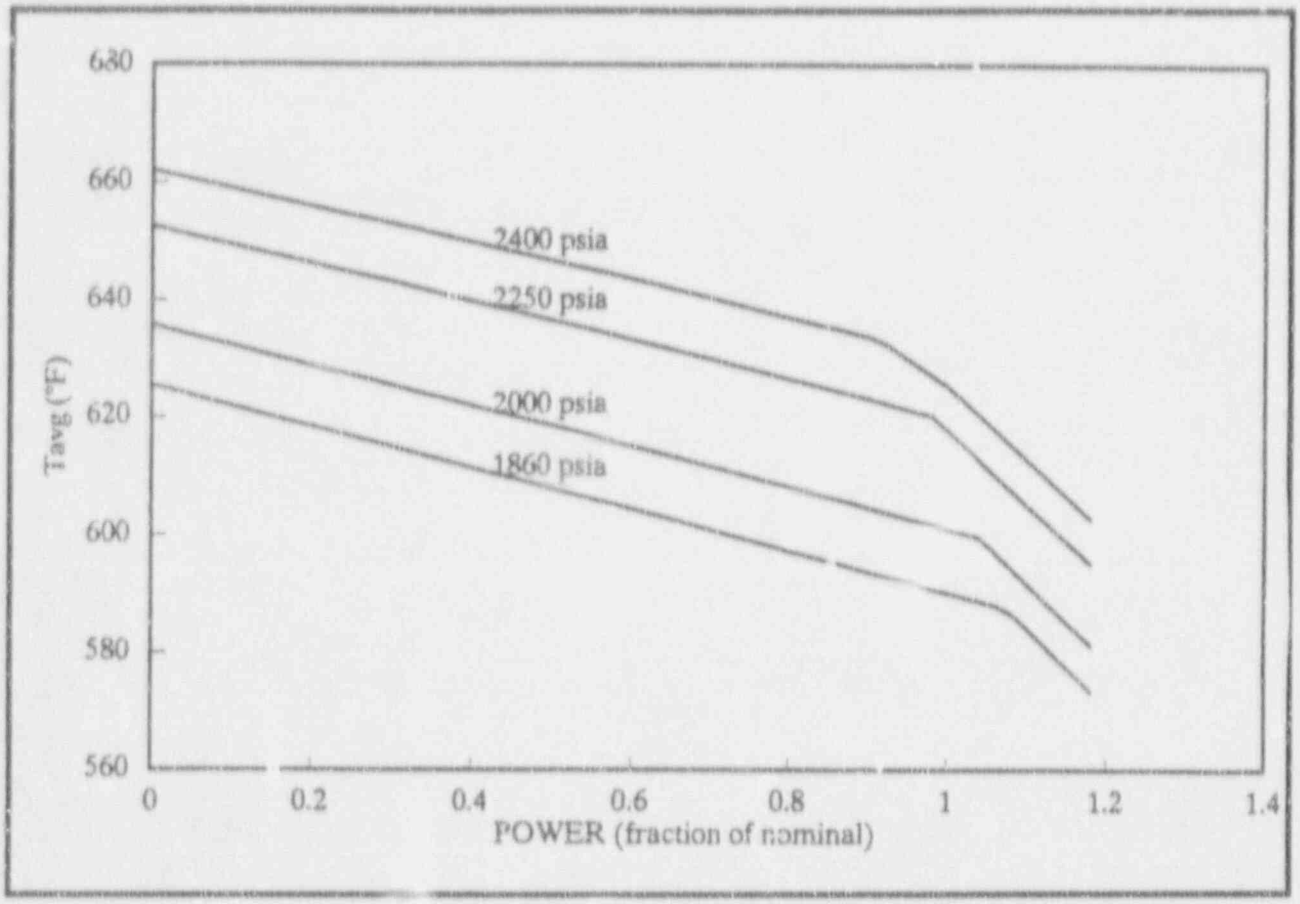


Figure 2.1-1a REACTOR CORE SAFETY LIMITS FOR THREE LOOP OPERATION FOR THE PERIOD OF OPERATION UNTIL STEAM GENERATOR REPLACEMENT

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - ≤ 25% of RATED THERMAL POWER High Setpoint - ≤ 103%** of RATED THERMAL POWER	Low Setpoint - ≤ 26% of RATED THERMAL POWER High Setpoint - ≤ 110%*** of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant > 2 seconds
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.3×10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	≥ 1870 psig	≥ 1860 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93% of instrument span
12. Loss of Flow	≥ 90% of design flow per loop *	≥ 89% of design flow per loop *

* Design flow per loop is one-third of the minimum allowable Reactor Coolant System Total Flow Rate as specified in Table 3.2-1.

** The high trip setpoint for Power Range, Neutron Flux, shall be ≤ 103% RATED THERMAL POWER for the period of operation until steam generator replacement.

*** The allowable value for the high trip setpoint for Power Range, Neutron Flux, is required to be ≤ 104% RATED THERMAL POWER for the period of operation until steam generator replacement.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 3 Loops	Operation with 2 Loops (no loops isolated) *	Operation with 2 Loops (1 loop isolated) *
K_1 = 1.2[illegible]4**	K_1 = ( )	K_1 = ( )
K_2 = 0.0220	K_2 = ( )	K_2 = ( )
K_3 = 0.001152	K_3 = ( )	K_3 = ( )

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between -44 percent and +3 percent, $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -44 percent, the ΔT trip setpoint shall be automatically reduced by 1.67 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds +3 percent, the ΔT trip setpoint shall be automatically reduced by 2.00 percent of its value at RATED THERMAL POWER.

* Values dependent on NRC approval of ECCS evaluation for these operating conditions.

** The value for K_1 shall be equal to 1.132 for the period of operation until steam generator replacement.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \left[K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S} \right) T - K_6 (T - T') - f_2(\Delta I) \right]$

Where:

ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
T	=	Average temperature, °F
T'	=	Indicated T_{avg} at RATED THERMAL POWER ≤ 586.8°F
K_4	=	1.079*
K_5	=	0.02/°F for increasing average temperature
K_5	=	0 for decreasing average temperatures
K_6	=	0.00164 for $T > T'$; $K_6 = 0$ for $T \leq T'$
$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
τ_3	=	Time constant utilized in the rate lag controller for T_{avg} $\tau_3 = 10$ secs.
S	=	Laplace transform operator (sec^{-1})
$f_2(\Delta I)$	=	0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2 percent span.

* The value for K_4 shall be equal to 1.016 for the period of operation until steam generator replacement.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 3 Loops in Operation	2 Loops in Operation ** & Loop Stop Valves Open	2 Loops in Operation ** & Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	≤ 591°F		
Pressurizer Pressure	≥ 2205 psig *		
Reactor Coolant System Total Flow Rate	≥ 284,000 gpm ***		

* Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10% RATED THERMAL POWER.

** Values dependent on NRC approval of ECCS evaluation for these conditions.

*** The value for the minimum allowable Reactor Coolant System Total Flow Rate is reduced to 268,500 gpm until steam generator replacement.

ATTACHMENT 2

DISCUSSION OF PROPOSED CHANGES

VIRGINIA ELECTRIC AND POWER COMPANY

Discussion of Proposed Changes

Background

North Anna Power Station Unit 1 is currently conducting a mid-cycle steam generator inspection outage. An extensive eddy current inspection of the North Anna Unit 1 steam generator tubes is being performed using very conservative analysis guidelines and plugging criteria. As such, a substantial number of tubes are expected to be plugged.

Based on projections of steam generator tube plugging levels, it was projected that the RCS total flow rate would not meet the current Technical Specifications minimum requirement of 284,000 gpm. Therefore, by letter dated January 8, 1992 (Serial No. 92-018), we proposed Technical Specifications changes to reduce the minimum total RCS flow rate to 275,300 gpm for the operating period until the steam generators could be replaced. This approximate 3% reduction in minimum RCS flow rate was intended to bound the effect of the increased flow resistance for the projected steam generator tube plugging levels.

According to Westinghouse estimates of RCS flow rates as a function of tube plugging percentage, the proposed 275,300 gpm minimum RCS flow rate corresponds to approximately 32% average steam generator tube plugging. Because measurement uncertainty may cause measured RCS flow rates to vary by as much as 2% from their true value, there exists an expected range of steam generator tube plugging over which the proposed flow rate may be met. This range is estimated to be between 28% and 36% average steam generator tube plugging. However, it should be emphasized that this is only an estimated range which assumes that the predictions of flow versus plugging are correct. If an inaccuracy in prediction is considered, the range of plugging over which the 275,300 gpm minimum measured flow might not be met is further widened.

As the steam generator tube inspections progressed, the adjustments to the tube plugging projections indicated that the "C" steam generator may exceed 30%. To accommodate the effect of additional tube plugging in the "C" steam generator, we requested a change to the North Anna Unit 1 operating license to limit the maximum reactor power level to 95% of rated thermal power for the interim period of operation until the steam generators could be replaced. This change was requested in our letter dated January 28, 1992 (Serial No. 92-042). The imposed 95% power restriction will provide sufficient margin in the large break Loss of Coolant Accident (LOCA) analysis to accommodate the interim effects of the increased steam generator tube plugging for the most restrictive steam generator.

In as much as the steam generator tube inspections are continuing and the actual tube plugging levels are not known, prudence requires us to cover the uncertainty in the range of possible tube plugging and flow rate measurement results. Therefore, we supplement our Technical Specification change request, submitted on January 8,

1992 (Serial No. 92-018), with a request to further reduce the minimum total RCS flow rate to 268,500 gpm. This change takes credit for the 5% power reduction discussed in our request for license amendment, submitted January 28, 1992 (Serial No. 92-042).

This Technical Specification change request supplements our January 8, 1992 submittal. The attached safety evaluation builds on the evaluation provided in our original submittal. Approval of this supplemental change is conditional on the prior approval of both the previously discussed change requests, i.e., the January 8, 1992 and the January 28, 1992 Technical Specification change submittals.

Introduction

As required by Technical Specifications 3.2.5 and 4.2.5.2, North Anna Unit 1 performs reactor coolant system (RCS) flow rate measurements subsequent to restart after each refueling. The North Anna Unit 1 safety analyses are based in part on verifying, via the Technical Specifications surveillance, that the Reactor Coolant System (RCS) total flow rate is greater than or equal to 284,000 gallons per minute (gpm). The additional steam generator tube plugging anticipated during the current mid-cycle inspection outage increases the likelihood of violating this Technical Specifications requirement. Therefore, safety analyses and evaluations have been performed which support this additional reduction in the RCS total flow rate limit to 268,500 gpm at 95% Rated Thermal Power. The attached safety evaluation has been prepared to support each of the Technical Specifications changes associated with this reduction in the RCS total flow rate limit.

Discussion of Proposed Changes

These proposed Technical Specifications changes supplement the changes discussed in our submittal dated January 8, 1992 in support of a reduced RCS total flow rate requirement to ≥268,500 gpm, which is an approximate 5-1/2% reduction from the current Technical Specifications requirement. These changes would only be effective until the North Anna Unit 1 steam generator replacement, which is currently scheduled to begin in January, 1993.

The proposed Technical Specifications changes affect Figure 2.1-1, Reactor Core Safety Limits (Specification 2.1.1), Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints (Technical Specification 2.2.1), and Table 3.2-1, DNB Parameters (Technical Specification 3.2.5).

The proposed Technical Specifications changes will revise Technical Specification 2.1.1 by placing a footnote on the bottom of the page referencing Figure 2.1-1a in lieu of Figure 2.1-1 for the Reactor Core Safety Limits. A revised reactor core safety limits graph (Figure 2.1-1a) is added with the inclusion of a new page 2-2a. The graph is changed to reflect the 268,500 gpm flow limit at 95% of Rated Thermal Power. Both the footnote and the graph title are worded to be effective for the period of operation until steam generator replacement.

The proposed change, as submitted January 8, 1992, requested a revision to the footnote on the bottom of Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints, to specify that the "design flow per loop" is "one-third of the minimum allowable Reactor Coolant System Total Flow Rate as specified in Table 3.2-1." Several additional items in Table 2.2-1 are proposed to be changed by this supplement. They are:

Item 2, Power Range, Neutron Flux, the high trip setpoint is lowered from 109% to 103% Rated Thermal Power and the allowable value for the high trip setpoint is lowered from 110% to 104% rated thermal power. Both changes are for the period of operation until steam generator replacement. These changes revise the trip setpoints so that credit can be taken for the 5% reduction in Rated Thermal Power level requested in our January 28, 1992 submittal. The revised setpoints will allow the necessary thermal margin to lower the total RCS flow rate to 268,500 gpm.

Item 7, Overtemperature ΔT, Note 1 (page 2-9), the value for K_1 is reduced from 1.264 to 1.132 for the period of operation until steam generator replacement. The change to this calculation factor revises the Overtemperature ΔT trip setpoints so that credit can be taken for the 5% reduction in Rated Thermal Power level and will ensure reactor protection with the lower the total RCS flow rate.

Item 8, Overpower ΔT, Note 2 (page 2-10), the value for K_4 is reduced from 1.073 to 1.016 for the period of operation until steam generator replacement. The change to this calculation factor revises the Overpower ΔT trip setpoints so that credit can be taken for the 5% reduction in Rated Thermal Power level and will ensure reactor protection with the lower the total RCS flow rate.

The remaining reactor trip setpoints specified in Table 2.2-1 will continue to ensure that the safety analysis assumptions will be met at the reduced RCS flow rate and Rated Thermal Power.

The proposed change, as submitted January 8, 1992, requested a revision to Table 3.2-1, DNB Parameters, by adding a footnote which reduces minimum limit for Reactor Coolant System Total Flow Rate from 284,000 gpm to 275,300 gpm until the North Anna Unit 1 steam generator replacement. With this supplemental change, we request to further lower the minimum limit for Reactor Coolant System Total Flow Rate to 268,500 gpm. This change takes credit for the 5% reduction in Rated Thermal Power level requested in our January 28, 1992 submittal. The proposed interim reduction in the minimum measured reactor coolant system flow is necessary to accommodate the expected increase in RCS loop resistance caused by increased steam generator tube plugging levels. Upon resumption of Cycle 9 power operation, the RCS total flow rate will be confirmed by measurement in accordance with Technical Specification 4.2.5.2.

The attached safety evaluation supports the above changes to the Technical Specifications. The supplemental changes to Specification 2.1.1, Figure 2.1-1, Table 2.2-1, and Table 3.2-1 are required on an interim basis until the steam generator replacement in 1993, at which time they will no longer apply.

ATTACHMENT 3

SAFETY EVALUATION

VIRGINIA ELECTRIC AND POWER COMPANY

TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 Background

North Anna Power Station Unit 1 is currently involved in a mid-cycle steam generator inspection outage. An extensive eddy current inspection of the North Anna Unit 1 steam generator tubes is being performed using very conservative analysis guidelines and plugging criteria. As such, a substantially increased number of tubes are expected to be plugged.

The physical consequences of extended SGTP are primarily (a) increased RCS loop resistance, resulting in a lower RCS flow rate, (b) decreased steam generator tube heat transfer area, resulting in lower steam generator outlet steam pressure, and (c) a decreased total RCS volume. The impact of these changes with respect to previously analyzed design conditions must be fully assessed for both normal operating and accident conditions. This assessment is performed following a steam generator inspection outage usually concurrent with a new reload safety evaluation. When required, revised safety analyses are performed and a Core Operating Limits Report (COLR) is prepared as required by Technical Specification 6.9.1.7.

In many cases, the incorporation of revised safety analyses into the North Anna design basis could be accomplished via Virginia Power processes employed to assess change per 10 CFR 50.59. However, based on current steam generator plugging projections, it is expected that the current North Anna 1 Technical Specification RCS flow limit (284,000 gpm) could be violated. This could potentially occur at average SGTP levels of

approximately 20%. To address this concern a separate Technical Specification Amendment request to reduce the RCS total flow rate limit by approximately 3% was submitted for review and approval (1,2).

1.2 Summary of Analyses and Evaluations to Date

In Reference 1, Virginia Power proposed a Technical Specification minimum measured flow of 275,300 gpm, which is approximately a 3% reduction from the currently licensed limit of 284,000 gpm. Because this flow rate is a critical input assumption in the UFSAR Chapter 15 analyses, it was necessary to evaluate all Chapter 15 analyses to support the implementation of extended SGTP at North Anna Unit 1. As a result, reanalyses of 5 UFSAR Chapter 15 non-LOCA accidents were presented (1). Accidents which were reanalyzed were:

Loss of External Load

Loss of Normal Feedwater

Rod Bank Withdrawal at Power

Complete Loss of Flow

Locked Reactor Coolant Pump Rotor

For the balance of accidents, evaluations were performed on the basis of parameter sensitivities and available thermal margins. These accident reanalyses and reevaluations were based on assumed operation at full rated thermal power. Supplemental information relating to this evaluation was provided in Reference 2.

Reference 3 presented a Large Break Loss of Coolant Accident (LBLOCA) reanalysis which supports operation of Unit 1 with maximum Steam Generator Tube Plugging (SGTP) level of up to 35% in any steam generator. That analysis was performed based on an assumed power level of 95% of rated thermal power.

1.3 Purpose of this Evaluation

This evaluation is being provided to supplement and extend References 1 and 2 to support a further reduction in RCS flow to 268,500 gpm. In this extension, operation at less than or equal to 95% of rated thermal power is assumed.

According to Westinghouse estimates of RCS flow rate as a function of tube plugging percentage, the Reference 1 proposed RCS flow rate of 275,300 gpm corresponds to approximately 32% average tube plugging. This estimate is based on an extrapolation of previous measured RCS flow data. Because RCS flow measurement uncertainty may cause measured flow rates to vary by as much as 2% from their true value, there exists an expected range of steam generator tube plugging over which the proposed flow rate may be met. This range is estimated to be between 28% and 36%. However, it should be emphasized that this is only an estimated range which does not explicitly account for inaccuracy in the prediction of flow versus plugging. If flow prediction accuracy is considered, the range of plugging over which the 275,300 gpm minimum measured flow rate requirement might not be met is further widened.

To provide for an increased confidence level that the Technical Specification flow limit will be met, Virginia Power has performed an additional evaluation which supports operation of North Anna Unit 1 at a thermal power level not to exceed 95% of rated thermal power, peak steam generator tube plugging levels of up to 35%, and a revised minimum reactor coolant system total measured flow rate of 268,500 gpm. Note that the Reference 3 submittal assumed a maximum power level of 95% of rated thermal power.

This evaluation provides an updated assessment of all of the UFSAR Chapter 15 accidents at the conditions stated in Table 1. To support operation at the 268,500 gpm measured flow rate, revised Core Thermal Limits (Technical Specifications Figure 2.1-1) have been developed. These revised limits have been used in turn to develop new Overtemperature and Overpower ΔT setpoints. Section 2.1.1 discusses this development. Consistent with established practice, confirmatory analyses of the uncontrolled rod withdrawal at power have been performed to verify DNB protection over a wide range of core thermal/hydraulic conditions. These analyses are discussed in Section 2.1.2.

The evaluations for the balance of the non-LOCA accidents are presented in Section 2.2. The impact of the reduced flow value on balance of plant and support systems is reviewed in Section 3.0. Conclusions and references are presented in Section 4.0 and 5.0, respectively.

TABLE 1

KEY EVALUATION ASSUMPTIONS

Initial Conditions

	Statistical DNB Method	Deterministic Method
Power	2748.35 MWt	2803.32 MWt
Average Temperature	586.8 °F	590.8 °F
RCS Flow Rate	268,500 gpm	263,130 gpm
Pressure	2250 psia	2220/2280 psia
$F_{\Delta h}$ at Assumed Power	1.512	1.573
1.55-Cosine Axial Power Profile		

2.0 EVALUATION

2.1 Impact of Flow Reduction on Core Thermal Limits

An evaluation has been performed to assess the impact of the proposed reduction in minimum measured flow rate on North Anna Unit 1 core thermal limits, Overtemperature and Overpower ΔT trip setpoints, and the f(ΔI) function.

The current Core Thermal Limits in Figure 2.1-1 of the Technical Specifications consist of two distinct limits. The DNBR portions of the limit lines are based on a minimum measured flow of 289,200 gpm and bound a design DNBR limit of 1.46 (as opposed to a statistical DNBR limit of 1.26). The vessel exit portions of the limit lines are based on a thermal design flow of 278,400 gpm.

Reference 1, an assessment was made of operation at a total measured RCS flow rate of 275,300 gpm and the current Core Thermal Limits. Operation was shown to be acceptable based on the use of available retained DNBR margins.

For the proposed additional reduction in minimum measured RCS flow rate to 268,500 gpm, revised thermal limit lines have been generated based on this reduced flow and a design DNBR limit of 1.46. In this way, no flow reduction penalty need be assessed against the generic retained DNBR margin for operation at 268,500 gpm. The definition and application of retained DNBR margin in Virginia Power design analyses is described in References 1 and 2.

The vessel exit boiling limited portions of the core thermal limits were evaluated with the proposed reduced non-statistical (deterministic) thermal design flow rate of 263,130 gpm (which corresponds to a minimum measured flow rate limit of 268,500 gpm). The revised Thermal Limits are shown in Figure 1.

FIGURE 1

CORE THERMAL LIMITS

Nominal T_{avg} = 586.8°F

Nominal RCS flow = 268,500 GPM

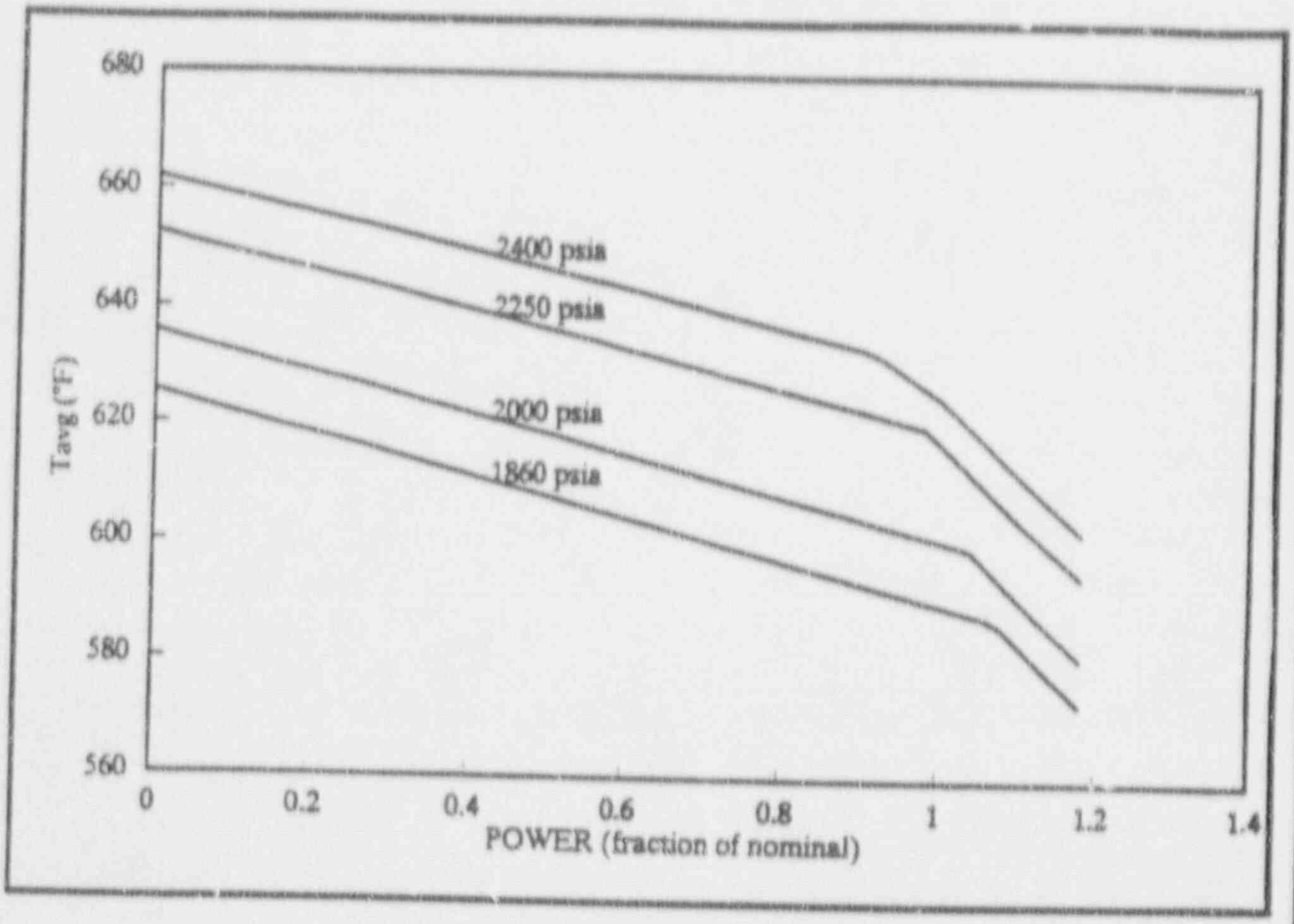


Figure 2.1-1a REACTOR CORE SAFETY LIMITS FOR THREE LOOP OPERATION FOR THE PERIOD OF OPERATION UNTIL STEAM GENERATOR REPLACEMENT

2.1.1 Generation of New OTΔT and OTΔT Setpoints

New Technical Specifications OTΔT and OPΔT setpoints were generated from the Figure 1 Thermal Limits using the Methodology of WCAP-8746 (Reference 4). The revised setpoint equations are compared to the existing equations in Tables 2 and 3.

For the OTΔT function, the f(ΔI) function currently in the Technical Specifications was demonstrated to provide bounding protection for the proposed reduced minimum measured flow rate. This evaluation was performed by the standard approach of demonstrating that the static power reduction associated with the f(ΔI) function maintains the DNBR above the design limit (1.46) for a range of positively and negatively skewed power shapes at the reduced flow rate.

Confirmatory analyses of the rod withdrawal at power accident were performed which demonstrate that the thermal limits are not exceeded over the entire range of achievable system conditions for operation with the revised protection setpoints. The results of these analyses are presented in the next section.

TABLE 2

Revised Overtemperature ΔT Setpoint Equation

$$\Delta T(\text{Setpoint}) = \Delta T(\text{Nom}) \times \left\{ K1 - K2 \frac{1+\text{tau1}s}{1+\text{tau2}s} (\text{Tave}-\text{T0}) + K3(P-P0) + f(\Delta I) \right\}$$

where	CURRENT	PROPOSED
K1 =	1.264	1.132
K2 =	0.0220	0.0220
K3 =	0.001152	0.001152
tau1 =	25 sec	25 sec
tau2 =	4 sec	4 sec
T0 =	586.8 °F	586.8 °F
P0 =	2235 psig	2235 psig
f(ΔI) =	0.0167x(-44%-ΔI), ΔI <-44%	No change
=	0.0, -44% <ΔI<+3%	No change
=	0.020x(ΔI-3%), ΔI > +3%	No change

Tave = Average temperature, °F
P = Pressurizer pressure, psig

TABLE 3

Revised Overpower ΔT Setpoint Equation

$$\Delta T(\text{Setpoint}) = \Delta T(\text{Nom}) \times \left\{ K4 - K5 \frac{\text{tau3}s}{1+\text{tau3}s} \text{Tave} + K6(\text{Tave}-\text{T0}) \right\}$$

where	CURRENT	PROPOSED
K4 =	1.079	1.016
K	0.02/°F, Tave increasing	0.02
[illegible]	0.0 Tave decreasing	0.0
K6	0.00164	0.00164
tau3 =	10 sec	10 sec
T0 =	586.8 °F	586.8 °F

f(ΔI) = 0.0, all ΔI

Tave = Average temperature, °F

2.1.2 Conf[illegible] [illegible] of Uncontrolled Rod Withdrawal at Power

The Rod Withdrawal at Power (RWAP) Accident has been reanalyzed to assess [illegible] minimum measured flow of 268,500 gpm associated with [illegible]. The [illegible]sed [illegible] setpoints described above were [illegible] analysis. [illegible] reduction in the high nuclear flux trip was [illegible] consistent with our proposed interim operation. A statistical treatment of key analysis uncertainties was utilized in accordance with North Anna implementation of the methodology described in Reference 5. A discussion of the analysis is presented in the following sections.

2.1.2.1 Accident Description

The uncontrolled rod cluster control assembly (RCCA) withdrawal at power is a postulated Condition II event initiated by operator action or control system malfunction. The transient is characterized by an increase in core heat flux, resulting in a mismatch between core power generation and power removal by the steam generator. This power mismatch, which persists until the steam generator pressure reaches the relief or safety valve setpoint, causes an increase in the primary coolant temperature. The transient would result in violation of the core thermal limits if not terminated by either manual or automatic action. The reactor protection system is designed to terminate the transient prior to exceeding core thermal limits.

2.1.2.2 Method of Analysis

The rod withdrawal at power event was reanalyzed with the RETRAN (6) system transient analysis code. All assumptions were consistent with or conservative with respect to those in the previously approved analyses. The RETRAN code provided transient pressures, core inlet temperatures, heat fluxes and core flows which were used as input to a detailed thermal/hydraulic statepoint analysis using the COBRA (7) code. The WRB-1 critical heat flux correlation (8) was used.

To fully evaluate the RWAP event, a wide range of initial plant conditions are analyzed to determine those which are most limiting. Previous analyses showed that the transients initiated from hot full power were limiting (1). Therefore the revised setpoints were reconfirmed by analyzing a range of reactivity insertion rates from an initial power level of 95% of rated thermal power.

It is assumed in the analysis that the steam dump and rod control systems do not function during the RWAP event. However credit is taken for pressurizer PORV's and safety valves, steam generator atmospheric relief valves and safety valves, as well as pressurizer spray (full flow from both valves is assumed), since studies have shown that this provides more limiting DNBR's.

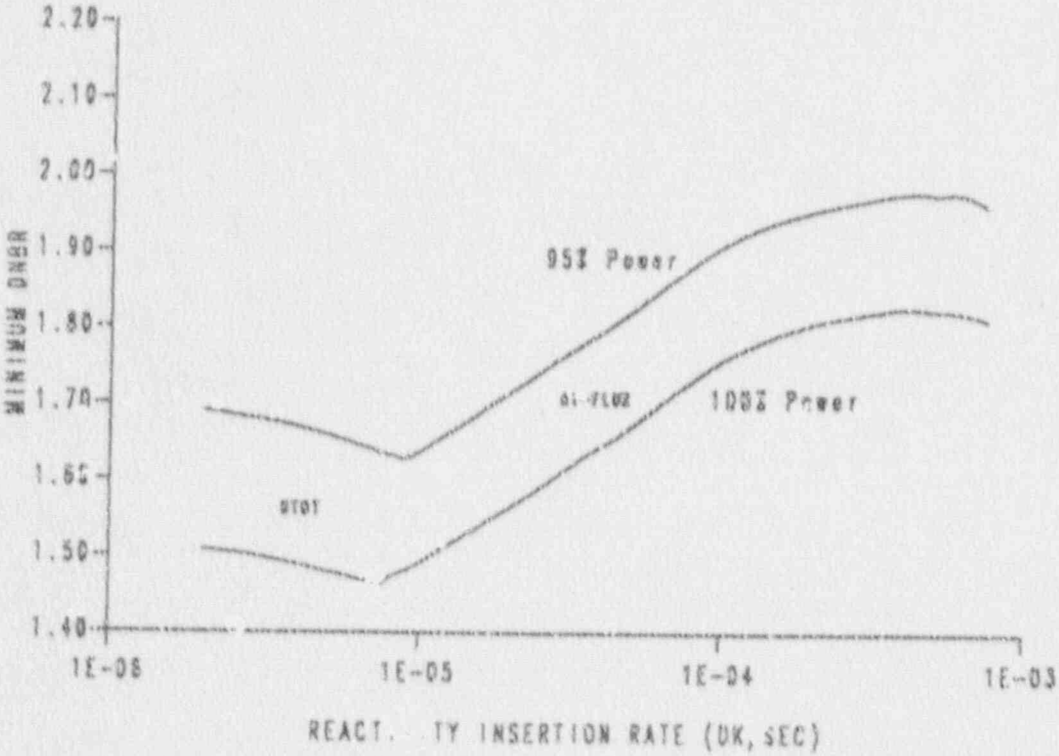
2.1.2.3 Results and Conclusions

The reanalysis of the rod withdrawal at power event demonstrated that the minimum DNBR will remain above the DNBR design limit for operation with reduced minimum measured flow associated with extended steam generator tube plugging.

Figure 2, on the following page, presents the minimum RWAP DNBR result as a function of reactivity insertion rate. The lower graph (labelled 100% Power) shows the Ref. 1 analysis results assuming the 275,300 gpm minimum measured flowrate, 100% power and the current Technical Specification OTΔT setpoints. The upper graph (labelled 95% power) shows the results from the revised analysis with reduced minimum measured flowrate, 95% power, the proposed Technical Specification OTΔT setpoints, and a reduction in the power range high flux trip setpoint. These results demonstrate that the combination of Overtemperature ΔT and high flux reactor trips act together to provide core DNB protection over the range of acheivable thermal conditions.

FIGURE 2

Effect of Renctivity Insertion Rote on Minimum DNBR
100% Power vs 95% Power, Minimum F.edback



2.2 Summary of Accident Evaluations

2.2.1 Overview of Assessment Process

As discussed in Reference 2, the process of evaluating the accidents for the effects of reduced flow is aided by what is essentially a screening process which subjects the individual accidents to the following tests:

1) Is the accident impacted by neither RCS flow nor steam generator tube plugging? In some cases (e.g., waste gas decay tank rupture), there is no impact and thus the event need not be considered further.

2) Is the accident impacted by plugging but not by flow? These events (e.g. chemical and volume control system malfunction at power, which is sensitive to RCS volume but not flow) will be addressed under 10 CFR 50.59 to support unit restart with extended plugging but have not been addressed here since they are not impacted by the proposed RCS flow Technical Specification Change.

3) Is the accident impacted by RCS flow alone (i.e. and not by other tube plugging phenomena)? In some cases the dynamics of the event are not impacted by plugging effects, and the impact is limited to the direct effect of RCS flow on the DNBR. An example is accidental depressurization of the reactor coolant system. Accidents in this category were dispositioned via application of a generic DNBR evaluation which shows that the effects of a 5% power reduction more than offset the impact of

the proposed additional flow reduction (i.e. with respect to the Reference 1 proposed value). This evaluation is discussed in Section 2.2.2, below.

4) Is the accident potentially impacted by both RCS flow and steam generator tube plugging effects? These are accidents which, in addition to the direct flow effect on DNBR, may be sensitive to

a) steam generator hydraulic resistance (i.e. pressure drop)
b) steam generator heat transfer area and/or secondary side initial conditions
c) reactor coolant system volume
d) instrumentation effects (i.e. OTΔT trip)

Accidents in this category must be evaluated, not only for the direct DNBR effects, as in Category 3) above, but also for the additional dynamic effects.

2.2.2 Assessment of Net Flow/ Thermal Power Impact on DNBR

Accidents in Category 3 above have been assessed using a generic evaluation of the net effect on calculated DNBR's of the proposed (a) reduction in core thermal power to 95% of rated thermal power and the associated increase in allowable radial power peaking, and (b) reduction in thermal design flow from 275,300 gpm (91,767 gpm/loop) to 268,500 gpm (89,500 gpm/loop). The analyses presented in Reference (1) support the establishment of a 275,300 gpm minimum measured RCS Total Flow Rate.

To perform this evaluation, a series of thermal/hydraulic statepoints which represent normal operation and limiting accident conditions were perturbed to determine the DNBR effect of marginal changes in flow, power, and $F_{\Delta H}$. The following statepoints were considered:

1) Nominal design hot full power conditions.
2) The statepoint corresponding to minimum DNBR following the complete loss of RCS flow event.
3) The limiting DNBR statepoint for the uncontrolled RCCA withdrawal event.

The worst-case sensitivities developed from each of these statepoints were then used to perform a conservative overall assessment of the net effect of a reduction in design flow from 275,300 gpm (the Reference 1 basis) to 268,500 gpm (approximately a 2.5% reduction) coincident with a 5% reduction in core thermal power and the associated increase in the

Technical Specification allowable $F_{\Delta H}$. The result was a net DNBR benefit ranging from 1.2-2.5%, depending on the statepoint examined.

2.2.3 Discussion of Accidents By Grouping

Group 1: No Flow/No Plugging Impact

Accidents which are impacted by neither flow nor plugging are those which are insensitive to RCS thermal/hydraulic conditions or have been demonstrated not to be credible (e.g. inactive loop start at power) for North Anna Unit 1. These events may be excluded from further consideration as discussed in Reference 1, and are as follows (UFSAR section number is provided below):

* Inactive Loop Startup (15.2.6)
* Misloaded Fuel Assembly (15.3.3)
* Waste Gas Decay Tank Rupture (15.3.5)
* Volume Control Tank Rupture (15.3.6)
* Fuel Handling Accident Outside Containment (15.4.5)
* Fuel Handling Accident Inside Containment (15.4.7)
* Steam Generator Tube Rupture (15.4.3)

As discussed above, the inactive loop start at power is not considered credible based on current Technical Specifications. RCS flow rate is not an analysis input parameter for the misloaded fuel assembly event, the tank rupture accidents or the fuel handling accidents. Further discussion of this category of events may be found in Reference (1).

Steam Generator Tube Rupture

The steam generator tube rupture event has been placed in this category because extended steam generator tube plugging and its accompanying effect on RCS flow, primary to secondary heat transfer, and RCS loop resistance would have insignificant impact on the analysis results of the steam generator tube rupture transient, as discussed in Reference 1.

Group 2: Accidents Impacted by Tube Plugging But Not RCS Flow Rate

Events which are insensitive to RCS flow but which can be impacted by SGTP levels are as follows (UFSAR section number is provided below):

* CVCS Malfunction (Boron Dilution) (15.2.4)
* Small Break LOCA (15.3.1)
* Large Break LOCA (15.4.1)

In general, these are events which are not assessed against the minimum DNBR criterion for moderate frequency events but rather by alternate criteria such as available operator response time (e.g. to avoid loss of shutdown margin for boron dilution events).

Boron Dilution

A reduction in the minimum measured flow rate has no direct consequences on the analysis of the boron dilution event. This has been discussed fully in Reference 1. The impact on the boron dilution at power analysis of the reduction in RCS volume associated with extended SGTP will be considered as part of the analysis supporting North Anna 1 Cycle 9 restart. Therefore, no reanalysis of this event is required to support the proposed Technical Specifications changes.

Large and Small Break LOCA

Reference 1 presented an assessment which concluded that both small and large break loss of coolant accident analysis results are insensitive to the proposed change in RCS flow rate. The prior evaluation's conclusions have been confirmed to remain applicable for the 268,500 gpm measured flow limit.

Group 3: Accidents Impacted by RCS Flow Alone

Accidents which are impacted by RCS flow only (and not by other tube plugging phenomena) are those where primary to secondary side heat transfer and/or steam generator primary side pressure drop, which are the two major impacts of SGTP apart from flow, do not impact the dynamics of the accident. For example, reactivity excursion transients which are rapid with respect to the thermal response time of the steam generators are placed into this category. Group 3 accidents in this category are (UFSAR section number is provided below).

* Control Rod Drop/Misalignment (15.2.3)
* Rod Withdrawal from Subcritical (15.2.1)
* Control Rod Ejection (15.4.6)

Control Rod Drop

The Virginia Power methodology for analysis of control rod drop was discussed in Reference 1. Reference 1 discussed the development of new dropped rod DNBR limit lines applicable to the 275,300 gpm total measured flow condition. These lines were based on a design DNBR limit which exceeded the statistical DNBR limit and therefore contained retained margin. This margin is more than enough to offset the reduction from the 275,300 gpm to the 268,500 gpm flow condition. Therefore the dropped rod limit lines Referenced in (1) are adequate to assess restart of Unit 1 at either the 275,300 gpm or the 268,500 gpm measured flow limit.

Rod Withdrawal From Subcritical

This event is assumed to be initiated from hot zero power. It is a non-limiting transient from a DNBR standpoint. The current analysis yields a peak heat flux well below the hot full power steady state value. The peak heat flux statepoint from the UFSAR analysis was reexamined at the proposed flow limit and the Technical Specifications radial peaking factor limit for hot zero power. The DNBR remains well above the limit and the conclusions of the UFSAR remain valid at the proposed flow limit for this event.

Control Rod Ejection

The control rod ejection evaluation in Reference 1 showed a small but acceptable increase in peak clad temperature for the limiting case based on application of flow sensitivity studies. Extending this assessment to the current proposed flow rate shows that the analysis limit for peak clad temperature is still met with substantial margin for the limiting case.

Other Events Assessed for DNB

For those events assessed against the DNBR criterion for moderate frequency events, the assessment in Section 2.2.2 applies. As noted therein, the impact of a 5% reduction in reactor power and the associated

increase in allowable $F_{\Delta H}$ has been shown to more than offset the impact of a 2.5% reduction in flow from the 275,300 gpm value previously assessed in Reference 1. Put another way, the assessments performed for 275,300 gpm design flow at 100% rated thermal power are applicable to 268,500 gpm design flow at 95% rated thermal power.

Group 3 accidents included in this category are as follows (UFSAR section number is provided below):

* Accidental Depressurization of the RCS (15.2.12)
* Excessive Load Increase (15.2.11)
* Excessive Heat Removal (15.2.10)
* Spurious Operation of the Safety Injection System (15.2.14)

Group 4: Accidents Potentially Impacted by RCS Flow and Plugging

Accidents potentially affected by both tube plugging and RCS flow are as follows (UFSAR section number is provided below):

* Partial Loss of Flow (15.2.5)
* Loss of External Load (15.2.7)
* Loss of Normal Feedwater (Loss of Offsite AC) (15.2.8/15.2.9)
* Accidental Depressurization of the Main Steam System (15.2.13)
* Minor Secondary Steam Pipe Breaks (UFSAR Section 15.3.2)
* Complete Loss of RCS Flow (15.3 4)
* Single Rod Withdrawal at Power (15.3.7)
* Major Secondary System Pipe Ruptures (Main Steam Line Break) (15.4.2.1)
* Rupture of a Main Feedwater Pipe (Main Feedline Break) (15.4.2.1)
* Locked Reactor Coolant Pump Rotor/ Sheared Shaft (15.4.4)

Partial Loss of Flow, Small Steam Pipe Breaks

The partial loss of flow event is bounded by the complete loss of flow event and its assessment is included within the scope of the evaluation for the more limiting event. Likewise the accidental depressurization of the main steam system and minor secondary steam pipe breaks are bounded by the main steam line break event and their assessments are included within the scope of the evaluation of the more limiting event.

Loss of External Electrical Load

The Loss of External Load event was reanalyzed for SGTP levels up to 40% and a reduced RCS minimum measured flow rate of 275,300 gpm (1). The analysis demonstrated that the transient does not challenge RCS and main steam system overpressure safety limits, nor does it approach DNB conditions.

The evaluation of Section 2.2.2 shows that the DNBR results of Reference (1) will remain bounding for the proposed condition. With respect to primary side overpressurization concerns, the reduction in RCS flow rate does not significantly affect primary or secondary side overpressurization results. Because the proposed revised design conditions also include a pe.. power level of 95%, the decreased load rejection would be expected to result in lower primary and secondary pressures than in the previous analysis.

It may be concluded that the effects of the proposed reduction in minimum measured RCS flow rate are bounded by the both the DNBR and overpressurization analysis results presented in Reference (1).

Loss of Normal Feedwater

The Loss of Normal Feedwater accident was reanalyzed as documented in Reference 1. The analysis demonstrated that steam generator tube plugging levels up to 40% (uniform) and a Tech Spec minimum measured RCS flow rate of 275,300 gpm do not adversely impact the ability of the auxiliary feedwater system to deliver adequate feedwater to prevent the relief of reactor coolant water through the pressurizer relief or safety valves, and to prevent system overpressurization. For the cases with and without continued operation of reactor coolant pumps, the feedwater flow rates required to provide adequate cooling were demonstrated to be well below actual deliverable pump flow rates.

Because this transient essentially evaluates the capability of the steam generators to remove core decay heat, the results of the analysis are primarily impacted by the level of SGTP rather than the reactor coolant system flow rate. The RCS flow rate impacts the nominal ΔT across the core, but does not significantly impact the steady state (or transient) removal of heat in such a long-term heat removal transient. The results of the loss of normal feedwater analysis therefore are insignificantly impacted by a further reduction of this minimum measured flow rate to 268,500 gpm. Furthermore, the 5% reduction in initial power level reduces both the initial stored energy and the post trip decay heat levels by an amount which more than offsets any small decrease in energy removal capability due to reduced RCS flow. Therefore the Reference 1 analysis remains bounding. Similar reasoning leads to the conclusion that

the Reference 1 evaluation for rupture of a main feedwater pipe remains bounding for the proposed condition.

Complete Loss of Flow

The complete loss of flow event has been reassessed for the proposed conditions by both applying the DNBR sensitivity studies discussed in Section 2.2.2 and examining the potential effects of higher loop resistance on the normalized RCS flow coastdown vs. time curve for this event. The latter effect was evaluated by assuming that the additional decrease in loop flow from Reference 1 to the proposed condition is entirely due to loop resistance effects. This effect was modelled in the flow coastdown analysis and the normalized (i.e. to time zero) coastdown curve was compared to the previous result. Based on this comparison and the DNBR sensitivities of Section 2.2.2, an estimate of the net effect of flow, power and radial peaking on minimum DNBR at the proposed condition was made. Note that the DNBR sensitivities developed in Section 2.2.2 enveloped the flow coastdown statepoint. It was concluded that the overall impact of the proposed flow limit at 95% power will result in a DNBR benefit with respect to the Reference 1 analysis. The Reference 1 analysis therefore bounds the proposed conditions.

Single Rod Withdrawal at Power

The Single Rod Withdrawal at Power event produces a system transient response which is similar to the uncontrolled control bank assembly withdrawal; that is, it results in an increase in core heat flux and a

mismatch between core power generation and power removal by the steam generators. This power mismatch, which persists until the steam generator pressure reaches the relief or safety valve setpoint, causes an increase in the primary coolant temperature. The transient would result in a violation of the analysis limits if not terminated by either manual or automatic action.

The reanalysis presented in Section 2.1.2 for the uncontrolled control bank withdrawal at power demonstrates that the proposed protection setpoints in combination with the proposed RCS flow limit provide adequate DNB protection for the full range of applicable thermal hydraulic conditions. The reload evaluation process demonstrates that less than 5% of the core will experience hot channel factors in excess of the steady state design limit for any single withdrawn RCCA. In this manner it can be demonstrated that less than 5% of the core will experience DNBR less than the design limit during a single RCCA withdrawal event, consistent with the conclusions of the UFSAR. This conclusion will be reconfirmed prior to Unit 1 startup.

Main Steamline Break

The main steamline break analysis is performed to demonstrate that there would be no core damage due to the onset of DNB, and that the energy release to containment does not cause failure of the containment structure.

Reference 1 demonstrated that the consequences of a main steamline break event under conditions of extended SGTP and reduced RCS flow rate would not exceed the analysis criteria as presented above. This was demonstrated on the basis that extended SGTP reduces the steam generator's capacity to remove energy from the RCS. Because the primary effect on the RCS of a main steamline break is to decrease RCS temperature and pressure, and to increase core power (given an end-of-cycle negative moderator temperature coefficient), a reduced capacity to remove energy from the RCS due to extended SGTP is an analysis benefit. It was concluded that that the calculated transient DNBR under conditions of extended steam generator tube plugging would be less limiting than the current licensing analysis.

For an additional reduction in RCS flow rate from 275,300 gpm to 268,500 gpm (a reduction of 2.5%), an additional penalty to be taken out of retained margin was developed. Utilizing a bounding flow sensitivity (+1.4% DNBR/% flow), the additional flow reduction translates to a 3.5% penalty to be assessed against available main steamline break analysis retained margin. This penalty fully accounts for the impact of the reduced RCS flow rate on MSLB DNBR analysis results.

<u>Locked Reactor Coolant Pump Rotor</u>

The locked reactor coolant pump rotor event was analyzed in Reference 1 for a flow rate corresponding to a measurement limit 275,300 gpm and hot full power operation. The results showed that less than 13% of the rods in the core experience DNBR less than the design limit, consistent

with the UFSAR. Also, the peak RCS pressure remained well within the acceptance limit.

For the case of an additional reduction of 2.5% in RCS flow and a 5% reduction in power, use of the DNBR sensitivities in Section 2.2.2 shows that the fraction of rods experiencing DNBR less than the design limit would be reduced. Also for the overpressure protection case, the effects of a 5% reduction in thermal power will more than offset the flow reduction effects. Since the dominant resistance to flow during the event is the locked rotor itself, the normalized (i.e. to time zero) flow coastdown curve for locked rotor is expected to remain essentially unchanged for the revised condition. The Reference 1 analysis remains bounding.

3.0 NSSS and Balance of Plant Systems and Components

3.1 NSSS Systems and Components

As documented in Reference 1, Westinghouse Electric Corporation performed reviews of the following NSSS components and systems to confirm that operation within the proposed conditions remains in compliance with the applicable codes and standards.

- Reactor Vessel and Internals
- Control Rod Drive Mechanisms
- Main Loop Isolation Valves
- Reactor Coolant Pump and Motor
- Pressurizer
- Steam Generator
- Auxiliary Systems Components (tanks, valves, heat exchangers)
- Fluid Systems
- Reactor Protection* and Control Systems

*See Section 2.1 for an assessment of new protection setpoints for the proposed conditions.

A review of the Reference 1 evaluation shows that the Westinghouse studies envelope operation at the proposed design flow rate of 268,500 gpm.

3.2 Balance of Plant Systems and Components

Stone and Webster Engineering Corporation has evaluated the effects of operating North Anna Unit 1 with reduced RCS flow and extended SGTP upon the balance of plant systems and components. The changes of significance for this assessment involve reductions in RCS flow, RCS volume, steam temperature and steam pressure. Engineering evaluations have been performed to demonstrate that these parameter changes and

resulting effects on plant systems and components will be bounded by existing analyses and will continue to meet applicable design criteria.

These major balance of plant design areas were evaluated (1):

- Accident Analyses
- Balance of Plant (BOP) Systems and Components
- Class I Piping
- Electrical Distribution System

A review of the Reference 1 assessment shows that the evaluation bounds the proposed operation at 95% of rated thermal power with an RCS total flow rate of 268,500 gpm.

4.0 CONCLUSIONS

A review of the accident analyses presented in UFSAR Chapter 15 has demonstrated that a reduction in minimum measured flowrate for North Anna Unit 1 to 268,500 gpm at 95% of rated thermal power is accommodated by current thermal margins or by the assessment of a penalty against available retained DNBR margins for all accidents. Explicit reanalyses were performed for the rod withdrawal at power event to confirm the adequacy of proposed revisions to the Overtemperature ΔT and high flux trip setpoints to be implemented for the balance of Unit 1 Cycle 9 operation. In addition, the remaining Engineered Safety Features and Reactor Protection System setpoints set forth in the Unit 1 Technical Specifications have been demonstrated to provide adequate plant protection at the reduced flow condition.

The evaluation showed that all of the acceptance criteria previously established in the UFSAR continue to be met for all of the events analyzed in Chapter 15. This conclusion will be reinforced by continued verification that core physics characteristics for operation with a reduced RCS flowrate remain within the envelope established by the reload safety evaluation to be performed prior to resumption of Cycle 9 operation.

The proposed temporary Core Thermal Limits have been verified to bound Unit 1 Cycle 9 operation with the new RCS flow rate.

A review of the NSSS design transients; NSSS fluid and control systems; reactor control and protection systems; NSSS primary components (including thermal and structural effects); and steam generator thermal/hydraulic performance has been performed. It was concluded that NSSS systems and components will continue to meet applicable acceptance criteria for operation with the reduced design flow rates and the associated steam generator tube plugging levels.

An engineering evaluation has also been performed to assess the impact of reduced flow and tube plugging on the existing containment integrity analyses (including the impact on Net Positive Suction Head-NPSH of engineered safeguards pumps) and containment subcompartment integrity analyses. The existing analyses were shown to remain bounding.

A balance of plant systems review shows continued acceptable performance under the reduced RCS flow/ extended tube plugging condition.

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