ENCLOSURE 1

LICENSING SUBMITTAL CHANGE PAGES

This enclosure contains change pages for the VCSNS licensing submittal of May 20, 1988 which modify the text and Technical Specifications markups to be appropriate for a design Tave of 587.4°F and a thermal design flow of 283500 gpm. All changes are highlighted by dual lines in the right margin. The following pages of the May 20th licensing submittal are affected.

ATTACHMENT	ITEM
1	Pages 5, 22, 25, 26, 31, 41, and 46
2	Figure 2.1-1, Safety Limits Table 2.2-1, Minimum Measured Flow Table 2.2-1, (Notes 1 & 3), Overpower-delta-T/Overtemperature-delta-T Trips Figure 3.2-2, RCS Flow Limits Figure 3.2-1, Tave Limit
6	Table 1

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2.0 SUMMARY AND CONCLUSIONS

Consistent with the Westinghouse standard reload methodology for analyzing cycle specific reloads, Reference 5, parameters were selected to conservatively bound the values for each subsequent reload cycle and to facilitate determination of the applicability of 10CFR50.59. The objective of subsequent cycle specific reload safety evaluations will be to verify that applicable safety limits are satisfied based on the reference evaluation/analyses established in this report. The mechanical, thermal and hydraulic, nuclear, and accident evaluations considered the transition core effects described for a VANTAGE 5 mixed core in Reference 1. The summary of these evaluations for the V. C. Summer core transitions to an all VANTAGE 5 core are given in the following sections of this submittal.

The transition design and safety evaluations consider the following conditions: 2775 MWt core thermal power, 555 °F core inlet temperature, 2250 psia system pressure and 283,500 gpm RCS thermal design flow. These conditions are used in core design and safety evaluations to justify safe operation with the conservative assumptions noted in Section 1.0. The conditions summarized in the SER for the VANTAGE 5 reference core report, WCAP-10444, have been considered in the V. C. Summer plant-specific safety evaluations.

The results of evaluation/analysis described herein lead to the following conclusions:

1. The Westinghouse VANTAGE 5 reload fuel assemblies for the V. C. Summer Nuclear Plant are mechanically compatible with the current LOPAR fuel assemblies, control rods, secondary source rods and reactor internals interfaces. The VANTAGE 5/LOPAR fuel assemblies satisfy the current design bases for the V. C. Summer reactor.

2. Evaluations/analyses have shown that all or any combination of thimble plugs may be removed from the Cycle 5 core and subsequent reload cores.

the safety analyses and the design DNBR values is broken down as follows. A fraction of the margin is utilized to accommodate the transition core penalty (12.5% for VANTAGE 5 fuel and none for LOPAR fuel) and the appropriate fuel rod bow DNBR penalty, Reference 10, which is less than 1.3%. The existing 6.3% margin in the LOPAR fuel and 17.5% margin in the VANTAGE 5 fuel between the design and safety analysis DNBR limits also includes a greater than 4% DNBR margin in the LOPAR fuel and a greater than 2.7% DNBR margin in the VANTAGE 5 fuel reserved for flexibility in the design.

The LOPAR and VANTAGE 5 designs have been shown to be hydraulically compatible in Reference 1.

The major impact of thimble plug removal on the thermal-hydraulic analysis is the increase in bypass flow which is reflected in Table 5.1.

The phenomena of fuel rod bowing, as described in Reference 10, must be accounted for in the DNBR safety analysis of Condition I and Condition II events for each plant application. Internal to the fuel rod, the IFBA and fuel pellet designs are not expected to increase the propensity for fuel rods to bow. External to the VANTAGE 5 fuel rod, the Inconel non-mixing vane and Zircaloy mixing vane grids provide fuel rod support. Additional restraint is provided with the Intermediate Flow Mixer (IFM) grids. Applicable generic credits for margin resulting from retained conservatism in the evaluation of DNBR are used to offset the effect of rod bow. The safety analysis for the V. C. Summer Plants maintain sufficient margin between the safety analysis limit DNBRs and the design limit DNBRs to accommodate full-flow and low-flow DNBR penalties.

The Westinghouse transition core DNB methodology is given in References 2 and 17 and has been approved by the NRC via Reference 18. Using this methodology, transition cores are analyzed as if they were full cores of one assembly type (full LOPAR or full VANTAGE 5), applying the applicable transition core penalties. This penalty is included in the safety analysis limit DNBRs such that sufficient margin over the design limit DNBR exists to accommodate the transition core penalty and the appropriate rod bow DNBR penalty.

TABLE 5.1 (Continued)

V. C. SUMMER THERMAL AND HYDRAULIC DESIGN PARAMETERS

HFP Nominal Coolant Conditions		Design Parameters
Vessel Minimum Measured Flow+		
Rate (including Bypass), 10^6 lbm/hr		108.1
GPM		289,500
Vessel Thermal Design Flow+		
Rate (including Bypass), 10^6 lbm/hr		105.9
GPM		283,500
Core Flow Rate		
(excluding Bypass, based on TDF)		
10^6 lbm/hr		96.47
GPM		258,210
Fuel Assembly Flow Area++		
for Heat Transfer, ft^2	(LOPAR)	41.55
	(V-5)	44.04
Core Inlet Mass Velocity,		
10^6 lbm/hr-ft (Based on TDF)	(LOPAR)	2.32
	(V-5)	2.19

\+ Includes 15% steam generator tube plugging

++ Assumes all LOPAR or VANTAGE 5 core

TABLE 5.1 (Continued)

V. C. SUMMER THERMAL AND HYDRAULIC DESIGN PARAMETERS

Thermal and Hydraulic Design Parameters		Design Parameters
(Based on Thermal Design Flow)		
Nominal Vessel/Core Inlet Temperature, °F		555.0
Vessel Average Temperature, °F		587.4
Core Average Temperature, °F		592.3
Vessel Outlet Temperature, °F		619.8
Average Temperature Rise in Vessel, °F		64.8
Average Temperature Rise in Core, °F		70.4
Heat Transfer		
Active Heat Transfer Surface Area,++ ft^2	(LOPAR)	48,598
	(V-5)	46,779
Average Heat Flux, BTU/hr-ft^2	(LOPAR)	189,820
	(V-5)	197,200
Average Linear Power, kw/ft		5.45
Peak Linear Power for Normal Operation,+++ kw/ft		13.30

++ Assumes all LOPAR or VANTAGE 5 core

+++ Based on 2.45 F_Q peaking factor

Reactor Coolant System Flow Reduction

All non-LOCA safety analyses reanalyzed for this report have incorporated a reduction in the reactor coolant system flow. The reduced flow corresponds to a thermal design flow of 283500 gpm and a minimum measured flow of 289500 gpm.

Thimble Plug Deletion

The non-LOCA analyses performed incorporated the impact of thimble plug deletion. Thimble plug deletion affects core pressure drops and bypass flow. These effects have been conservatively incorporated into the non-LOCA safety analyses.

Debris Filter Bottom Nozzle

The VANTAGE 5 fuel design will also include the Debris Filter Bottom Nozzle (DFBN). In the DFBN, the relatively large flow holes in the conventional bottom nozzle are replaced with a new pattern of smaller flow holes. These holes are sized to minimize the passage of debris particles large enough to cause damage while still providing sufficient flow area, comparable pressure drop, and continued structural integrity of the nozzle. As such, no parameters important to the non-LOCA safety analyses are impacted.

Increased Overpower/Overtemperature ΔT Reactor Trip Response Time

The total time delay of the overtemperature ΔT and overpower ΔT trips (including RTD time response, trip circuitry and channel electronics delay) assumed in the non-LOCA analyses is 8.5 seconds. The 8.5 second delay includes a 7 second first order lag incorporated into the determination of the time at which the overtemperature ΔT and overpower ΔT trip setpoints are reached. The remaining 1.5 seconds is the delay from the time at which the trip signal is initiated until the rod cluster control assemblies are free to drop into the core.

7.0 SUMMARY OF TECHNICAL SPECIFICATION CHANGES

The proposed changes to the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications are summarized in Table 7.1. These changes reflect the impact of the design, analytical methodology, and safety analysis assumptions outlined in the SCE&G amendment request and are given in the proposed Technical Specification page changes (see Attachment 2 of this report). A brief overview of the significant changes follows.

7.1 Core Safety Limits

Core safety limits and associated bases for 3-loop operation during modes 1 and 2 are revised to reflect the impact of the transition to VANTAGE 5 with:

1. The use of ITDP and the WRB-1 and WRB-2 DNB Correlation.

2. An $F_{\Delta H}$ of 1.62 (see Section 7.11).

3. Reduced RCS flow to accommodate the increased resistance of the VANTAGE 5 fuel assembly and to support SG tube plugging up to 15% (see Section 7.2).

The proposed limits corresponds to those for the LOPAR fuel which are limiting during the transition period. Less limiting values will be possible with a full core of VANTAGE 5.

7.2 Thermal Design Flow

The VCSNS thermal design flow is being decreased from 288600 gpm to 283500 gpm. This flow reduction accommodates:

a. The increased resistance of the VANTAGE 5 fuel assembly.

b. Up to 15% SG tube plugging in all three SG's.

7.11 Nuclear Enthalpy Rise Hot Channel Factor

The following $F_{\Delta H}$ values (includes uncertainties) are proposed for the VANTAGE 5 transition.

$$F_{\Delta H} = 1.56 [1 + 0.3 (1-P)]$$

where P is the fraction of full power. These higher values allow increased fuel cycle design flexibility and lower leakage core loading patterns.

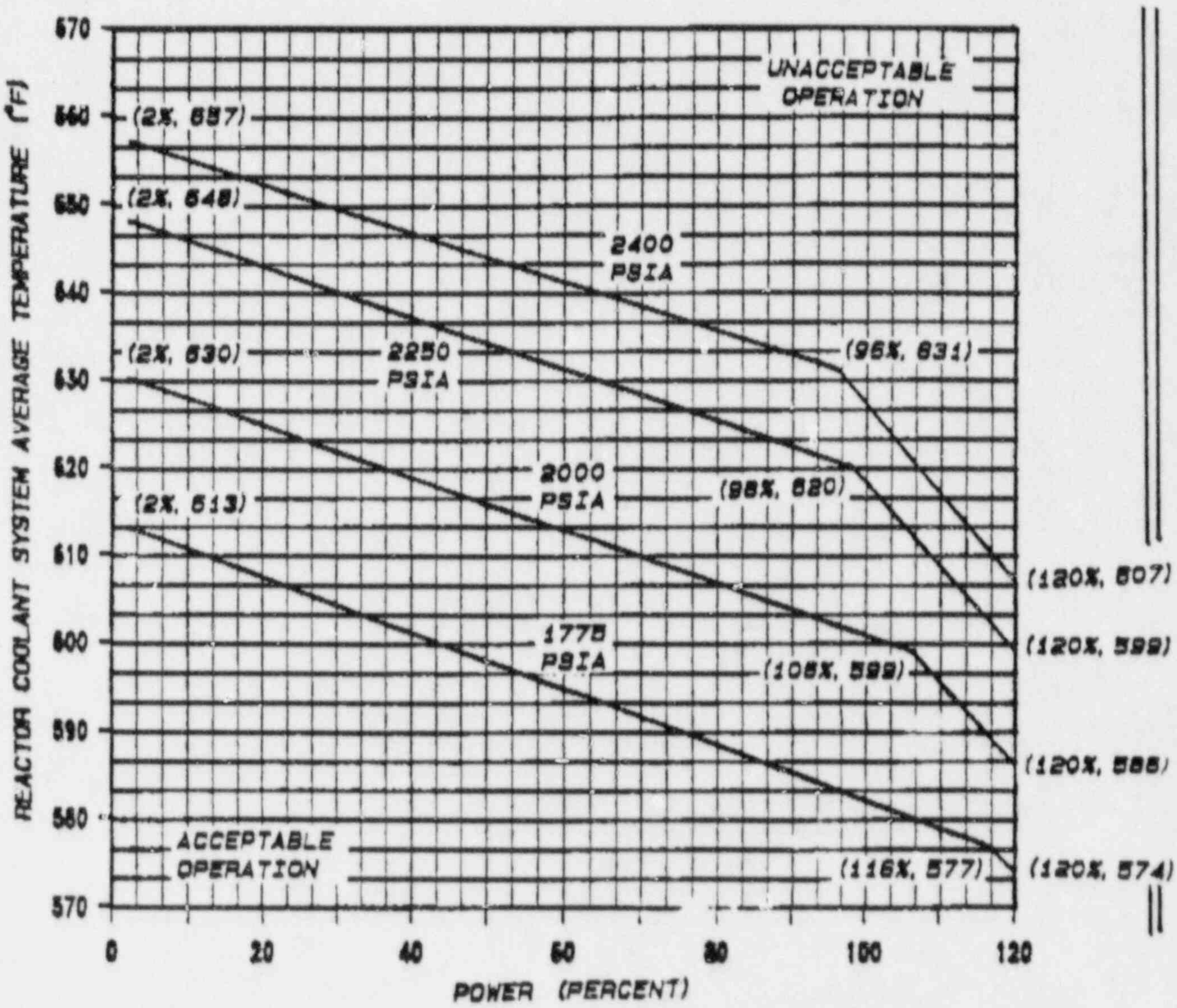
7.12 DNB Parameters

The proposed limits on DNB related parameters (T_{avg} and Pressurizer Pressure) assure that each are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The proposed revisions are consistent with new accident analyses supplied in the Transient Safety Evaluation which utilizes the ITDP (see Section 5.0) for DNB evaluations.

The T_{avg} reflects the nominal baseline T_{avg} of 587.4°F assumed in the VANTAGE 5 analysis in order to support full power operation with:

1. 15% uniform SG tube plugging.

2. A thermal design flow conservatively calculated to support 15% SG tube plugging and the use of VANTAGE 5 fuel.



* When operating in the reduced RTP region of Technical Specification 3.2.3 (Figure 3.2-3) the restricted power level must be considered 100% RTP for this figure.

Figure 2.1-1 Reactor Core Safety Limit - Three loops in Operation

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	S	Trip Setpoint	Allowable Value
1. Manual Reactor Trip	Not Applicable	NA	NA	NA	NA
2. Power Range, Neutron Flux High Setpoint	7.5	4.56	0	≤109% of RTP	≤111.2% of RTP
Low Setpoint	8.3	4.56	0	≤25% of RTP	≤27.2% of RTP
3. Power Range, Neutron Flux High Positive Rate	1.6	0.5	0	≤5% of RTP with a time constant ≥2 seconds	≤6.3% of RTP with a time constant ≥2 seconds
4. Power Range, Neutron Flux High Negative Rate	1.6	0.5	0	≤5% of RTP with a time constant ≥2 seconds	≤6.3% of RTP with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.4	0	≤25% of RTP	≤31% of RTP
6. Source Range, Neutron Flux	17.0	10.0	0	≤10^5 cps	≤1.4 x 10^5 cps
7. Overtemperature ΔT	9.8 ~~7.1~~	7.29 ~~2.94~~	1.9 & 1.2** ~~1.8~~	See note 1	See note 2
8. Overpower ΔT	5.2 ~~4.5~~	2.26 ~~1.4~~	1.9 ~~1.2~~	See note 3	See note 4
9. Pressurizer Pressure-Low	3.1	0.71	1.5	≥1870 psig	≥1859 psig
10. Pressurizer Pressure-High	3.1	0.71	1.5	≤2380 psig	≤2391 psig
11. Pressurizer Water Level-High	5.0	2.18	1.5	≤92% of instrument span	≤93.8% of instrument span
12. Loss of Flow	2.5	1.0	1.5	≥90% of loop design flow*	≥89.2% of loop design flow*

* Loop design flow = ~~96,200~~ 96,500 gpm
RTP = RATED THERMAL POWER

** 1.9% SPAN FOR DELTA-T (RTDS) AND 1.2% FOR PRESSURIZER PRESSURE

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \leq \Delta T_0 \{K_1 - K_2 \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} [T - T'] + K_3(P - P') - f_1(\Delta I)\}$$

Where:

ΔT = Measured ΔT by RTD Manifold Instrumentation

ΔT_0 $\leq$ ~~=~~ Indicated ΔT at RATED THERMAL POWER

K_1 $\leq$ ~~=~~ ~~1.090~~ 1.203

K_2 $\geq$ ~~=~~ ~~0.01450~~ 0.03006

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1, & τ_2 = Time constants utilized in the lead-lag controller for T_{avg}, τ_1 $\geq$ ~~=~~ 28 secs., τ_2 $\leq$ ~~=~~ 4 secs.

T = Average temperature °F

T' $\leq$ ~~587.4~~ ~~585.5~~ 587.4°F Reference T_{avg} at RATED THERMAL POWER

K_3 $\geq$ ~~=~~ ~~.0006728~~ 0.00147

P = Pressurizer pressure, psig

P' $\geq$ ~~=~~ 2235 psig, Nominal RCS operating pressure

S = Laplace transform operator, sec^{-1}.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

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 ~~-35~~ -24 percent and ~~+8~~ 4 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 ~~-35~~ -24 percent, the ΔT trip setpoint shall be automatically reduced by ~~1.62~~ 2.27 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds ~~+8~~ 4 percent, the ΔT trip setpoint shall be automatically reduced by ~~1.11~~ ~~2.14~~ 2.13 percent of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 2.0 ~~3.4~~ percent ΔT span.

NOTE 3: OVERPOWER ΔT

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

Where: ΔT = as defined in Note 1

ΔT_o = as defined in Note 1

K_4 $\leq$ ~~=~~ ~~1.08~~ 1.0875

K_5 $\geq$ ~~=~~ 0.02/°F for increasing average temperature and 0 for decreasing average temperature

$\frac{\tau_3 S}{1 + \tau_3 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

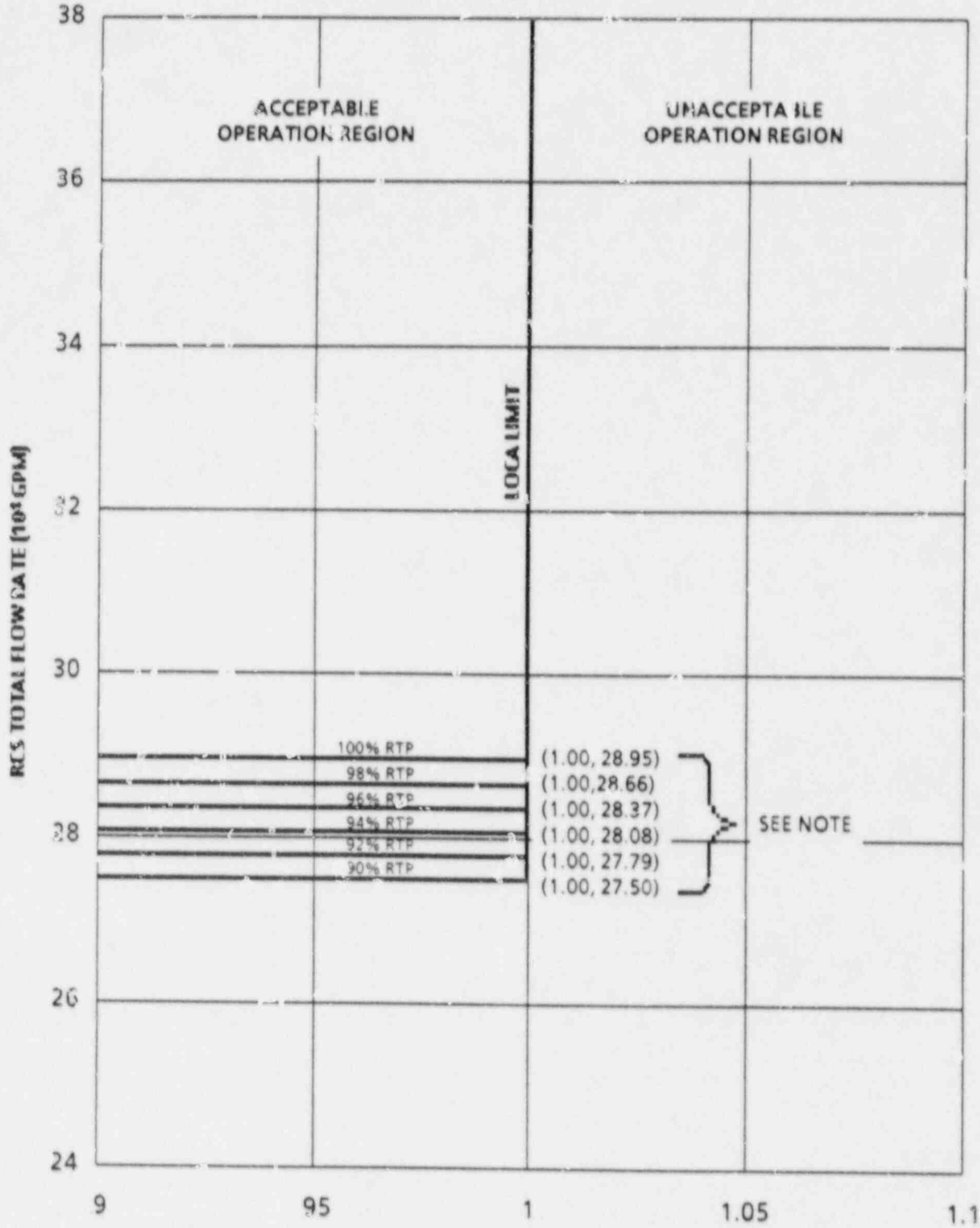
NOTATION (Continued)

NOTE 3: (Continued)

τ_3	=	Time constant utilized in the rate-lag controller for T_{avg}, τ_3 ~~[illegible]~~ ≥ 10 secs.
K_6	~~[illegible]~~ ≥	0.~~001190~~ 0.00156/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$
T	=	as defined in Note 1
T''	≤	~~[illegible]~~ 587.4°F Reference T_{avg} at RATED THERMAL POWER
S	=	as defined in Note 1

NOTE 4: The channel's maximum trip setpoint shall not exceed its computed trip point by more than ~~[illegible]~~ 2.0 percent ΔT span.

MEASUREMENT UNCERTAINTIES OF 2.1% FOR FLOW AND 4.0% FOR INCORE MEASUREMENT OF $F^N_{\Delta H}$ ARE INCLUDED IN THIS FIGURE



$R = F^N_{\Delta H}/1.56\ [1.0 + 0.3(1.0\text{-}P)]$

FIGURE 3.2-3 RCS TOTAL FLOW RATE VS. R THREE LOOP OPERATION

NOTE: When operationg in this region, the restricted power levels shall be considered to be 100% of rated thermal power (RTP) for Figure 2.1-1

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 3 Loops In Operation	LIMITS: 2 Loops in Operation
Reactor Coolant System T_{avg}	~~589.8°F~~ $\leq$ ~~592°F~~ 591.7°F	**
Pressurizer Pressure	$\geq$ ~~2220 psia*~~ 2206 psia*	**

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

**These values left blank pending NRC approval of two-loop operation.

Table 1

VCSNS ANALYSIS BASELINE

Parameter	Current Value	Proposed Value For Vantage 5 Transition
NSS Power, MWt	2785	2787
Core Power, MWt	2775	2775
System Pressure, psia	2250	2250
Thermal Design Flow, gpm	288600	283500
Core Bypass Flow, %	6.4	8.9**
T_{AVE}, °F	587.4	587.4
T_{HOT}, °F	618.7	619.8
$F_{\Delta H}$	1.55	1.62
$F_{\Delta H}$ Multiplier	0.2	0.3
LOCA F_Q	2.25	2.45
SG Tube Plugging, %	16	15
AFD Control	CAOC	RAOC
Peaking Surveillance	F_{xy} (z)	F_Q (z)
High Head Safety Injection	Recirculation Isolated	Recirculation Not Isolated
Thimble Plugs	Yes	Optional

** Non-ITDP