ATTACHMENT 1

MARKED-UP TECHNICAL SPECIFICATION PAGES

Page	Specification	Change Description
3/4 7-1	3.7.1.1, Turbine Cycle, Safety Valves	Delete **Action b.** Rename **Action c** as **Action b.**
3/4 7-2	3.7.1.1, Turbine Cycle, Safety Valves	Delete Table 3.7-2.
3/4 7-3	3.7.1.1, Turbine Cycle, Safety Valves	Reformat Table 3.4-3 per Insert 1. Also renumber Table 3.7-3 as 3.7-2 and move to page 3/4 7-2
B 3/4 7-1	3/4.7.1.1 Bases, Turbine Cycle	Delete references to two loop operations.
B 3/4 7-2	3/4.7.1.1 Bases, Turbine Cycle	Delete references to two loop operations.

9102120007 910204
PDR ADOCK 05000395
P PDR

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table ~~3.7-3.~~ 3.7-2

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With 3 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~b. With 2 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves associated with an operating loop inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~c.~~ b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	87
2	65
3	43

DELETE

~~TABLE 3.7-2~~

~~MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 2 LOOP OPERATION~~

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator*	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	**
2	**
3	**

~~*At least two safety valves shall be OPERABLE on the non-operating steam generator.~~

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

INSERT 1

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

S/G A	S/G B	S/G C		Lift Setting*		Orifice Size
XVS-2806A	XVS-2806F	XVS-2806K		1176 psig ±1%		4.515 In dia/16 sq in
XVS-2806B	XVS-2806G	XVS-2806L		1190 psig ±3%		4.515 In dia/16 sq in
XVS-2806C	XVS-2806H	XVS-2806M		1205 psig ±3%		4.515 In dia/16 sq in
XVS-2806D	XVS-2806I	XVS-2806N		1220 psig ±3%		4.515 In dia/16 sq in
XVS-2806E	XVS-2806J	XVS-2806P		1235 psig ±3%		4.515 In dia/16 sq in

*The Lift Setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REPLACE WITH INSERT 1 AND MOVE TO PAGE 3/4 7-2

TABLE 3.7-3

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LIFT SETTING ([illegible]%)*	ORIFICE SIZE
S/G A		
XVS-2806 A	1176 psig	4.515 In dia/16 sq in
XVS-2806 B	1190 psig	4.515 In dia/16 sq in
XVS-2806 C	1205 psig	4.515 In dia/16 sq in
XVS-2806 D	1220 psig	4.515 In dia/16 sq in
XVS-2806 E	1235 psig	4.515 In dia/16 sq in
S/G B		
XVS-2806 F	1176 psig	4.515 In dia/16 sq in
XVS-2806 G	1190 psig	4.515 In dia/16 sq in
XVS-2806 H	1205 psig	4.515 In dia/16 sq in
XVS-2806 I	1220 psig	4.515 In dia/16 sq in
XVS-2806 J	1235 psig	4.515 In dia/16 sq in
S/G C		
XVS-2806 K	1176 psig	4.515 In dia/16 sq in
XVS-2806 L	1190 psig	4.515 In dia/16 sq in
XVS-2806 M	1205 psig	4.515 In dia/16 sq in
XVS-2806 N	1220 psig	4.515 In dia/16 sq in
XVS-2806 P	1235 psig	4.515 In dia/16 sq in

*The Lift Setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 13.76×10^6 lbs/hr which is 110 percent of the total secondary steam flow of 12.2×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table
37-1 3.7-2.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 3 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times 109$$

~~For 2 loop operation~~

$$\cancel{SP = \frac{(X) - (Y)(U)}{X} \times (*)}$$

Where:

SP = Reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = Maximum number of inoperable safety valves per steam line

~~U = Maximum number of inoperable safety valves per operating steam line~~

SAFETY VALVES (Continued)

109 = Power Range Neutron Flux-High Trip Setpoint for 3 loop operation.

~~* = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for 2 loop operation. This value left blank pending NRC approval of 2 loop operation.~~

X = Total relieving capacity of all safety valves per steam line in lbs/hour.

Y = Maximum relieving capacity of any one safety valve in lbs/hour.

3/4.7.1.2 EMERGENCY FEEDWATER SYSTEM

The OPERABILITY of the emergency feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

The emergency feedwater system is capable of delivering a total feedwater flow of 380 gpm at a pressure of 1211 psig to the entrance of at least two steam generators while allowing for (1) any spillage through the design worst-case break of the emergency feedwater line, (2) the design worst-case single failure, and (3) recirculation flow. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F at which point the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 11 hours with steam discharge to the atmosphere concurrent with total loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

ATTACHMENT 2

DESCRIPTION OF AMENDMENT REQUEST

SAFETY EVALUATION

DESCRIPTION OF AMENDMENT REQUEST

This amendment request involves Technical Specification (T.S.) 3/4.7.1.1 "Turbine Cycle - Safety Valves" and addresses two separate changes. The first change is strictly administrative in nature, and the second change is a request to modify the acceptable setpoint tolerance associated with the Main Steam Safety Valves (MSSVs).

Presently, T.S. 3/4.7.1.1 contains provisions which were included for the NRC's eventual approval of two-loop power operations. These provisions consisted of Action Statement b. which specifically applies to two-loop operation, and table 3.7-2 which prescribes the maximum power allowed during two-loop operation based on the number of inoperable MSSVs. SCE&G is requesting the removal of these provisions based on the fact that it appears highly unlikely that two-loop operation will be approved. SCE&G is also concerned, from a human factors perspective, that having irrelevant matter in the specification could be detrimental to its application.

The second item to be addressed involves the setpoint tolerance for the MSSVs. The current Limiting Condition For Operation (LCO) requires that the MSSVs be operable with lift settings as specified in Table 3.7-3. It is important to note that there are five MSSVs per steam line, and that the setpoint of each of the five valves increases sequentially in increments of essentially fifteen psig (i.e., the lowest setpoint valve on each steam line is 1176 psig, the next is 1190 psig, the next is 1205, and so on). Table 3.7-3 currently imposes a ± 1% tolerance on all of the MSSV lift setpoints. SCE&G is requesting that the lift setpoint tolerance be increased from ±1% to ± 3% for the four highest set MSSVs per steam line while maintaining ±1% as the setpoint tolerance on the lowest set MSSV per steam line.

The operability of the MSSVs ensures that the maximum pressure experienced by the secondary system will be limited to 110% (1305 psig) of design pressure (1185 psig) during the most severe anticipated transient. T.S. 3/4.7.1.1 requires that the MSSVs be tested and verified operable in accordance with Section XI of the ASME Boiler & Pressure Vessel (B&PV) Code. The code does not contain a setpoint tolerance; therefore, the ±1% setpoint tolerance prescribed in T.S. 3.7.1.1 is applied as an acceptance criteria. SCE&G proposes to increase the setpoint tolerance to ±3% based on the advancement in technology which can more accurately determine the lift setpoint and the inability to make the corresponding <u>fine</u> adjustments to the MSSVs. Also, an evaluation of a ±3% tolerance shows that the related effects of a larger setpoint tolerance yields no safety concerns and does not prevent the MSSVs from performing their design function.

SAFETY EVALUATION:

An evaluation was performed to ensure that increasing the MSSV setpoint tolerance from ± 1% to ± 3% did not compromise safety. The impact of the tolerance change was assessed with respect to the following areas: The ASME B&PV Code, Westinghouse's Safety Analyses, Gilbert Commonwealth's Safety Analyses, and the Technical Specification Margin of Safety.

The MSSVs are in compliance with the ASME B&PV Code Section III (1971 edition, Winter 1972 addendum) which provides requirements for the design of the MSSVs. No requirements exist in Section III regarding the tolerance on the lift pressure setpoint. However, there is a requirement in Section III that the valve itself be designed to have a popping point tolerance of ± 1% (i.e., the repeatability of the valve is within ± 1%). The Inservice Inspection (ISI) required by T.S. 4.7.1.1 is in compliance with Section XI of the ASME B&PV Code ('77 ed., S'78 add.); also, this version of Section XI does not specify a setpoint tolerance requirement. However, the 1989 Edition of Section XI currently refers to ISI guidance which states that safety valves must not exceed their stamped set pressure by 3% or greater. Therefore, a MSSV setpoint tolerance of ± 3% does not contradict the B&PV Code currently committed to and is consistent with the most recent ISI guidance provided by the Code.

Westinghouse performed a safety evaluation to address increasing the setpoint tolerance of all the MSSVs from ± 1% to ± 3% with respect to its effects on the Reactor Coolant System (RCS) and the LOCA and non-LOCA licensing basis events. This evaluation included an analysis of each event that is discussed in Virgil C. Summer Nuclear Station (VCSNS) Reload Transition Safety Report (RTSR), an analysis of each FSAR LOCA related analysis. and an analysis of VCSNS's Steam Generator Tube Rupture Analysis. The results of the Westinghouse evaluation concluded that all licensing basis criteria continue to be met, and the conclusions in the RTSR remain valid.

Westinghouse also published a letter (CGE-90-1157) which, based on an examination of the VCSNS Licensing Basis Analyses, revealed that the maximum relief capacity required by the MSSVs to satisfy the most severe anticipated transient was 82.3% of rated steam flow. A subsequent letter (CGE-90-1160) verified that 82.3% total relief capacity can be substituted in lieu of the 110% value provided in section V-2 of the Westinghouse Steam Systems Design Manual. Provided with this information, Gilbert/Commonwealth (G/C)--the Architect Engineer for VCSNS Steam Systems--performed a calculation (G/C calculation DC-501-0428-11, Rev. 0) and verified that only four of the five MSSVs are required to meet the licensing basis events. Based on the results of their calculations, G/C performed an analysis of the increase in the MSSV's setpoint tolerance from ±1% to ±3% and concluded that the increased tolerance does not affect the MSSV's ability to perform their design function. However, it was discovered that the setpoint of the lowest set MSSV is used in the Emergency Feedwater (EFW) System capacity calculations. A change to ± 3% for this MSSV could affect the EFW system's capability to meet the Westinghouse criterion of a maximum 65%/35% split of EFW flow to any two steam generators. Therefore, the change in setpoint tolerance of the lowest set MSSV is not included in this amendment request.

Finally the Technicl Specification margin of safety was evaluated. The purpose for the MSSV's as described in the Bases is to limit the most severe anticipated transient to 110% of design pressure (1305 psig) and to maintain the lift settings and capacities consistent with Section III of the ASME B&PV Code, 1971 edition. Therefore, the T.S. margin of safety is the margin between 1305 psig and the pressure at which ultimate failure of the secondary pressure boundary occurs. As previously stated, an examination of the licensing bases for VCSNS has shown that changing the MSSV's setpoint tolerance from ±1% to ± % does not cause an increase in the maximum upset pressure and is consistent with Section III of the ASME B&PV Code, 1971 edition. Thus, the T.S. margin of safety is not affected by this change.

Based on an indepth review of the evaluations described above, SCE^ has concluded that the amendment request continues to meet the requirements of the ASME B&PV Code and involves no significant increase in safety consequences.

ATTACHMENT 3

DESCRIPTION OF AMENDMENT REQUEST

NO SIGNIFICANT HAZARDS EVALUATION

DESCRIPTION OF AMENDMENT REQUEST:

This amendment request involves Technical Specification (T.S.) 3/4.7.1.1 "Turbine Cycle - Safety Valves" and addresses two separate changes. The first change is strictly administrative in nature, and the second change is a request to modify the acceptable setpoint tolerance associated with the Main Steam Safety Valves (MSSVs).

Presently, T.S. 3/4.7.1.1 contains provisions which were included for the NRC's eventual approval of two-loop power operations. These provisions consisted of Action Statement b. which specifically applies to two-loop operation, and table 3.7-2 which prescribes the maximum power allowed during two-loop operation based on the number of inoperable MSSVs. SCE&G is requesting the removal of these provisions based on the fact that it appears highly unlikely that two-loop operation will be approved. SCE&G is also concerned, from a human factors perspective that having irrelevant matter in the specification could be detrimental to its application.

The second item to be addressed involves the setpoint tolerance for the MSSVs. The current Limiting Condition For Operation (LCO) requires that the MSSVs be operable with lift settings as specified in Table 3.7-3. It is important to note that there are five MSSVs per steam line, and that the setpoint of each of the five valves increases sequentially in increments of essentially fifteen psig (i.e., the lowest setpoint valve on each steam line is 1176 psig, the next is 1190 psig, the next is 1205, and so on). Table 3.7-3 currently imposes a $\pm 1\%$ tolerance on all of the MSSV lift setpoints. SCE&G is requesting that the lift setpoint tolerance be increased from $\pm 1\%$ to $\pm 3\%$ for the four highest set MSSVs per steam line while maintaining $\pm 1\%$ as the setpoint tolerance on the lowest set MSSV per steam line.

The operability of the MSSVs ensures that the maximum pressure experienced by the secondary system will be limited to 110% (1305 psig) of design pressure (1185 psig) during the most severe anticipated transient. T.S. 3/4.7.1.1 requires that the MSSVs be tested and verified operable in accordance with Section XI of the ASME Boiler & Pressure Vessel (B&PV) Code. The code does not contain a setpoint tolerance; therefore, the $\pm 1\%$ setpoint tolerance prescribed in T.S. 3.7.1.1 is applied as an acceptance criteria. SCE&G proposes to increase the setpoint tolerance to $\pm 3\%$ based on the advancement in technology which can more accurately determine the lift setpoint and the inability to make the corresponding <u>fine</u> adjustments to the MSSVs. Also, an evaluation of a $\pm 3\%$ tolerance shows that the related effects of a larger setpoint tolerance yields no safety concerns and does not prevent the MSSVs from performing their design function.

No Significant Hazards Determination:

This amendment request has been reviewed with respect to Title 10 of the Code of Federal Regulations (10CFR) part 50.92 and found to contain no significant hazards considerations for the following reasons:

1) The amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated. The effects of the requested change was examined with respect to each event described in the RTSR (non-LOCA events), the small and large break LOCA accidents, and the Steam Generator Tube Rupture Event. The examination revealed that the conclusions reached for all events described in the RTSR remained valid and the results of the FSAR accident analyses were not impacted.

2) The amendment request does not create the possibility of a new or different kind of accident from any accident previously evaluated. The requested change does not represent a design change in that all design limits are maintained and the physical design of all systems are unaffected. Therefore, the potential for malfunction or failure of any component or system as a result of the requested change remains unaffected.

3) The amendment request does not involve a significant reduction in a margin of safety. The requested change does not affect the minimum or maximum pressures experienced by the main steam system during any licensing basis event and remains consistent with the margin of safety as described in the bases of the Technical Specifications.

Again, for the reasons listed above and supported by the attached safety evaluation (Attachment 4), SCE&G has determined that the requested amendment to T.S. 3/4.7.1.1 has no significant hazards considerations.

ATTACHMENT 4

SUPPORTING DOCUMENTS

10CFR50.59 evaluation supporting an increased Main Steam Safety Valve lift pressure setpoint tolerance of $\pm 3\%$.

VIRGIL C. SUMMER NUCLEAR STATION
10CFR50.59 SAFETY EVALUATION WORKSHEET

Check Applicable Yes [ ] and No [ ] Indications

PARENT DOCUMENT ______________

Does this evaluation change the Final Safety Analysis Report or Fire Protection Evaluation Report?

*1

Yes [ ]   No [X]

TECH. SPEC. REFERENCE	
Section	Page
3.7.1.1	3/4 7-3
TABLE 3.7-3	
	B 3/4 7-1

Not addressed in Tech Specs [ ]

FSAR/FPER REFERENCE		
Chapter	Section	Page
10	3.2.4	10.3-4

Not addressed in FSAR/FPER [ ]

Is a change in Tech. Specification involved?

*2

Yes [X]   No [ ]

Nuc. Lic. Reviewer / Date

PSRC/NSRC Review

Request and Receive Nuclear Regulatory Commission Authorization For Change Prior To Implementation Of the Subject Change

Authorization Denied

Abort The Change

UNREVIEWED SAFETY QUESTION DETERMINATION

Answer the seven questions on pages 2 and 3. Provide specific reasons justifying the decision for the "yes" or "no" answers.

NOTE:

Restatement of the question in a negative sense or making a simple statement of conclusion is not sufficient and shall be avoided. It is recognized, however, that for certain very simple activities, a statement of the conclusion with identification of references consulted to support the conclusion will be adequate.

Complete the items below after the questions on pages 2 and 3 have been addressed.

*3

Any Answer Yes [X]   All Answers No [ ]

Authorization Received

Initiate The Change

*If answer (1) is "yes" but answers (2) and (3) are "no", then the change is reportable under 10CFR50.59b and a description of the change will be included in the Annual Report. If answer (2) is "yes" then 10CFR50.59 is not applicable. Proceed to 10CFR50.90

G. D. Williams Jr. — 1/21/91
Lead Engineer/Preparer — Date

[signature] — 1-29-91
Independent Reviewer — Date

[signature] — 1/30/91
Approval Signature — Date

South Carolina Electric & Gas Co. wishes to submit a Tech. Spec. change to increase the set point tolerances of the four (4) highest set Main Steam Safety Valves (MSSV) on each loop from +/- 1% to +/- 3%. The lowest set MSSV on each loop will remain the same (+/- 1%). The following paragraphs show that the Tech. Spec. margin of safety for the MSSV set point tolerances has not been reduced.

PLANT TECHNICAL SPECIFICATIONS - MSSV's

During normal operation, Tech. Spec. 3.7.1.1 (Table 3.7-3) currently requires that the MSSV's be set as follows:

MAIN STEAM SAFETY VALVES			MSSV SET POINT +/- 1%
MAIN STEAM LOOP A	MAIN STEAM LOOP B	MAIN STEAM LOOP C	
XVS-2806 A	XVS-2806 F	XVS-2806 K	1176 PSIG
XVS-2806 B	XVS-2806 G	XVS-2806 L	1190 PSIG
XVS-2806 C	XVS-2806 H	XVS-2806 M	1205 PSIG
XVS-2806 D	XVS-2806 I	XVS-2806 N	1220 PSIG
XVS-2806 E	XVS-2806 J	XVS-2806 P	1235 PSIG

A visit to the Bases for this particular Tech. Spec. indicates that the MSSV's are required to operate to prevent the secondary system pressure from exceeding 110% (1305 psig) of its design pressure (1185 psig). The Bases for this Tech. Spec. state that the valve lift settings and relieving capacities are in accordance with ASME B&PV Code Section III requirements. The ASME code requires that anticipated transient pressures cannot exceed 110% of design pressure. Since design pressure is 1185 psig (1200 psia) then 110% of design pressure is 1305 psig (1320 psia). Hence the Tech. Spec. secondary system limit is the same as that required by the ASME Code. The ASME Code margin of safety consists of the area between 110% of design pressure and the pressure which causes ultimate failure of the pressure boundary. In this case, the ASME Code margin of safety and thus the real Tech. Spec. margin of safety is the area between 1305 psig and ultimate failure of the secondary system pressure boundary.

ASME B&PV CODE

The ASME B&PV Code Section III ('71 ed., W'72 add.) under subsection NC-7411 requires that the "total rated relieving capacity shall, ..., be sufficient to prevent a rise in pressure of more than 10 per cent above system design pressure ... under any pressure transients anticipated to arise". NC-7511 further requires that at least one safety valve be set at the system design pressure. NC-7512 requires that pressure drop including back pressure be considered in meeting the 110% of design pressure requirement. No mention is made of any requirement to consider any safety valve set point tolerance in the system design. NC-7614.3 requires that the safety valve itself shall have a popping-point tolerance of +/- 1%. However, Section XI ('77 ed., S'78 add.) does not specify a set point tolerance.

OM-1 specifies that the valves when tested shall not exceed their stamped set pressure by 3% or greater. The corresponding ASME B&PV Code Edition & Addenda for Section III subsection NC-7000 do not address the 3% tolerance for valves with set pressures over 1000 psig.

WESTINGHOUSE SAFETY ANALYSES

Current plant design incorporates 5 MSSV's on each loop with a combined name plate capacity of 110% of full rated flow at 100% reactor power. The set points on the MSSV's for each loop are staggered in banks. Each bank consists of one valve on each loop (three valves total, all having the same set point). Each bank (five total) has a capacity of 22% of full rated flow.

The Westinghouse safety analyses for anticipated transients is bounded by the "Loss of Load / Turbine Trip @ 100% Power" event for over-pressurization events. This event requires a capacity of 82.3% of full rated flow in order to keep maximum secondary system pressure below 110% of design pressure. Therefore, the valves which have a combined capacity of 82.3% are all that is required to meet the ASME code requirements and thus meet Tech. Spec. requirements. Since, four banks of valves have a combined capacity of 88%, the fifth bank of valves is not required.
NOTE: <u>It should be noted that the difference between the Westinghouse safety analyses requirements (82.3% full rated flow)</u>

and the ASME / Tech. Spec. requirements is design margin **not** Tech. Spec. margin.

The following paragraphs address the specific safety analyses looked at by Westinghouse. It is a synopsis of Westinghouse Nuclear Safety Evaluations No.'s SECL 89-939 and 1140. No technical basis has been changed.

Historically, the 1% tolerance of the Pressurizer Safety Relief Valve (PSRV) and the Main Steam Safety Valve (MSSV) set points has been negligible with respect to the safety analyses: and thus, has not been accounted for. However, an increase in the tolerance to +/- 3% is considered to be sufficiently significant such that its impact on the safety analyses should be considered.

Modifying either side of the tolerance band potentially affects the safety analyses. The PSRV's and MSSV's provide protection from over-pressurization of the primary and secondary systems, respectively. By increasing the positive side of the tolerance band, the pressure at which the safety valves potentially lift and thus the potential maximum pressure attained is increased. By increasing the negative side of the tolerance band, the pressure at which the safety valves potentially lift is decreased.

A Tech. Spec. change has previously been submitted to and approved by the NRC to increase the PSRV set point tolerance to +/- 3%. As a result, this evaluation conservatively assumes that the valve lift set points for both the PSRV's and the MSSV's are increased to +/- 3%. Furthermore, it is assumed that the accumulation point for the PSRV's and the MSSV's occurs at a pressure 3% above the actual valve lift set point. This is more conservative than the ASME code requirement which states that the accumulation point occur within 3% above the nominal valve lift set point for the valve.

NON-LOCA

Each non-LOCA licensing event is discussed below in the order in which it appears in the Reload Transition Safety Report (RTSR) for the Virgil C. Summer Nuclear Station (VCNS).

1. Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal from a Subcritical Condition (RTSR Section 15.2.1)

For this condition II event, rod withdrawal results in a rapid reactivity insertion and increase in core power potentially leading to high local fuel temperatures and heat fluxes and a reduction in the minimum DNBR. The transient is promptly terminated by a reactor trip on the Power Range High Neutron Flux - low set point. Due to the inherent thermal lag in the fuel pellet, heat transfer to the RCS is relatively slow and the minimum DNBR is shown to remain above the limit value. No credit is taken for the MSSV's. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

2. Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal at POWER (RTSR Section 15.2.2)

For this condition II event, various initial power levels and reactivity insertion rates for both minimum and maximum feedback assumptions are analyzed. The resulting power excursion may lead to high local fuel temperatures and heat fluxes and a reduction in the minimum DNBR. Since this event is a limiting DNB event and not peak pressure limiting, the Pressurizer PORV's are conservatively assumed to be operable. Neither the primary nor the secondary systems reach the reduced safety set point during this event. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

3. Rod Cluster Control Assembly Misoperation (RTSR Section 15.2.3)

This condition II event is analyzed to demonstrate that following various RCCA misoperation events such as dropped rod(s)/bank or statically misaligned rods, that the minimum DNBR remains above the limit value. Neither the primary nor the secondary systems reach the reduced safety set point during this event. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

4. Uncontrolled Boron Dilution (RTSR Section 15.2.4)

This condition II event is analyzed for all six modes of operation. This analysis demonstrates that sufficient negative reactivity exists, such that, should a dilution event occur, there is sufficient time following an alarm to allow operator detection and termination of the event prior to a complete loss of shutdown margin and return to criticality. The Mode 1 dilution analysis is bounded by the RCCA withdrawal at power event (RTSR 15.2.2, see item 2) while the Mode 2 dilution analysis continues to be bounded by the RCCA withdrawal at hot zero power (RTSR 15.2.1, see item 1). The MSSV set point relaxation for these events has already been addressed. For the dilution analyses performed in Modes 3 through 6, since adequate operator action time is assured prior to reaching criticality, no additional heat is added to the core and no pressurization of the primary or secondary systems occurs. Changes in the MSSV set point tolerances will have no effect on the calculated available operator action time. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

5. Partial Loss of Forced Reactor Coolant Flow (RTSR Section 15.2.5)

This condition II event is analyzed under full power conditions assuming that 1 of 3 operating reactor coolant pumps coasts down. The reactor is promptly tripped on low reactor coolant loop flow. The analysis demonstrates that the minimum DNBR remains above the limit value. The RCS pressure increases above the initial value during the event yet never reaches the reduced safety valve set point. The MSSV's are not actuated during the simulation of this event. Note that no credit is taken for the observed RCS pressure rise in the DNB analysis. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

6. Startup of an Inactive Reactor Coolant Loop (RTSR Section 15.2.6)

This condition II event is analyzed assuming a maximum initial power level consistent with 2 loop operation and the P-8 set point. The startup of an inactive loop results in a reactivity insertion since the inactive loop fluid is at a lower temperature than the rest of the core. The analysis demonstrates that the minimum DNBR remains above the limit value. The RCS pressure increases above the initial value yet never reaches the reduced safety valve set point. The MSSV's are not actuated during the simulation of this event. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

7. Loss of External Electrical Load and/or Turbine Trip (RTSR Section 15.2.7)

The analysis presented in the RTSR represents a complete loss of steam load from full power without a direct reactor trip. Four cases are analyzed, maximum and minimum feedback, with and without pressure control. The analysis demonstrates that, with the power mismatch between the core and turbine, the primary and secondary system pressures remain below 110% of design and that the minimum DNBR remains above the limit value. A sensitivity analysis was performed using the LOFTRAN computer code assuming the PSRV and MSSV characteristics discussed in the introduction. The peak pressurizer pressure was calculated to be 2636 psia for the minimum feedback without pressure control case. The peak secondary pressure was calculated to be 1271 psia for all four cases. Thus both the primary and secondary pressures continue to remain below 110% of design and the minimum DNBR continues to remain above the limit value. Should the MSSV's actuate at a pressure 3% lower than nominal, adequate relief capacity exists to prevent over-pressurization of the secondary side. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

8. Loss of Normal Feedwater (RTSR Section 15.2.8)

The analysis presented in the RTSR represents a complete loss of feedwater from full power. The loss of the secondary side heat sink results in a heatup and pressurization of the primary and secondary systems. The analysis demonstrates that adequate emergency feedwater flow is delivered to the steam generators to remove decay heat such that over-pressurization of the primary and secondary systems will not occur and the pressurizer does not fill. Should the MSSV's actuate at a lift set point up to 3% below nominal, the maximum secondary and primary side temperatures will be beneficially reduced. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

9. Loss of Offsite Power to the Station Auxiliaries (Station Blackout) (RTSR Section 15.2.9)

The analysis presented in the RTSR represents a complete loss of power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc., from full power. The loss of power results in a heatup and pressurization of the primary and secondary systems. The analysis demonstrates that adequate emergency feedwater flow is delivered to the steam generators to remove decay heat such that DNB will not occur, and the pressurizer does not fill. Should the MSSV's actuate at a lift set point up to 3% below nominal, the maximum secondary and primary side temperatures will be beneficially reduced. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

10. Excessive Heat Removal Due to Feedwater System Malfunctions (RTSR Section 15.2.10)

The analysis presented in the RTSR illustrates the plant response to a 250 % step increase in the feedwater flow to one steam generator from full power, and a step increase in feedwater flow to one steam generator at zero power. The analysis demonstrates that from zero power the reactivity transient, and thus the minimum DNBR, is bounded by the rod withdrawal from sub critical event. For the full power case, the minimum DNBR is shown to remain above the limit value. The MSSV's are not actuated during this event even if the MSSV lift set point is reduced by up to 3%. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

11. Excessive Load Increase Incident (RTSR Section 15.2.11)

The analysis presented in the RTSR describes plant response to a 10% step increase in load. Four different cases are analyzed: minimum and maximum feedback, with and without reactor control. For each case it is shown that the minimum DNBR remains above the limit value. The cases which assume no reactor control result in an RCS depressurization as the heat extraction from the secondary side increases. The cases which take credit for reactor control result maintain the RCS pressure at essentially the initial value. Since an increase in load results in a secondary side pressure reduction the MSSV's are not actuated. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

12. Accidental Depressurization of the Reactor Coolant System (RTSR Section 15.2.12)

For this Condition II event, the transient is initiated by the opening of a single pressurizer relief or safety valve at full power. Initially, the RCS pressure drops rapidly until pressure reaches the hot leg saturation pressure. At this time the pressure decrease continues but at a slower rate. The analysis demonstrates that the minimum DNBR remains above the limit value. This event does not pressurize the secondary side. As a result, the MSSV's are not challenged. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

13. Accidental Depressurization of the Main Steam System (RTSR Section 15.2.13)

For this Condition II event, the transient is initiated by the full opening of a single steam dump, relief, or safety valve at zero power. The analysis confirms that the minimum DNBR remains above the limit value. Since the secondary side pressures drop immediately following initiation of the event, the MSSV's are not actuated. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

14. Spurious Operation of the Safety Injection System at Power (RTSR Section 15.2.14)

For this Condition II event, a spurious Safety Injection Signal (SIS) is assumed to be generated at full power. The injection of borated water into the RCS reduces core power, temperature, and pressure until the reactor trips on low pressurizer pressure. The power and temperature reduction causes a similar reduction in pressure on the secondary side. Since the secondary side pressures drop immediately following initiation of the event, the MSSV's are not actuated. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

15. Minor Secondary Side Pipe Breaks (RTSR Section 15.3.2)

This Condition III event continues to be bounded by the analysis presented in RTSR Section 15.4.2 (see items 19 and 20 below).

16. Inadvertent Loading of a Fuel Assembly into an Improper Position (RTSR Section 15.3.3)

For the event presented in the RTSR, the loading of a fuel assembly into an improper position would affect the core power shape. Since the power shape and not the total power generated would be affected, the steam system conditions will remain unaffected such that the MSSV's would not be affected. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

17. Complete Loss of Forced Reactor Coolant Flow (RTSR Section 15.3.4)

This Condition III event is analyzed under full power conditions assuming 3 of 3 operating reactor coolant pumps coast down. The reactor is assumed to trip on an undervoltage signal. The analysis demonstrates that the minimum DNBR remains above the limit value. In the DNB analysis, no credit is taken for the increase in pressure. The RCS pressure increases above the initial value during the event yet never reaches the safety valve set point. The MSSV's are not actuated during this event. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

18. Single Rod Cluster Control Assembly (RCCA) Withdrawal at Full Power (RTSR Section 15.3.6)

For this Condition III event, two cases are analyzed and presented in the RTSR: automatic and manual reactor control. In both cases an increase in core power, coolant temperature, and hot channel factor result in a reduction in the minimum DNBR. The analysis demonstrates that, although it is not possible for all cases to ensure that DNB will not occur, an upper bound on the number of fuel rods experiencing DNB is less than or equal to 5%. Since this event is a limiting DNB event and not peak pressure limiting, credit is not taken for any pressure increase associated with this event. The MSSV's are not actuated during this event. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

19. Rupture of a Main Steam Line (RTSR Section 15.4.2.1)

For this Condition IV event, the transient is assumed to be initiated by the instantaneous double-ended rupture of a main steam line. Since the secondary side pressures drop immediately following initiation of the event, the MSSV's are not actuated. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

20. Rupture of a Main Feedwater Pipe (RTSR Section 15.4.2.2)

For this Condition IV event, the double-ended rupture of a main feedwater pipe initially results in a cool down of the RCS due to the heat removal of the steam generator blowdown. This cool down period is followed by a heat up as the high levels of decay heat and the lack of inventory on the secondary side results in inadequate heat transfer. The event is analyzed to show that adequate heat removal capability exists to remove core decay heat and stored energy following a reactor trip from full power and that the core remains in a coolable geometry. This is accomplished by applying the strict criterion that no hot leg boiling occurs during the transient. For this event, the MSSV's are actuated during the heatup phase following reactor trip. A sensitivity analysis has been performed using the LOFTRAN code assuming the increased MSSV set points. Maximum steam system pressures were calculated to be 1272 psia. Minimum subcooling margin in the RCS was found to be 23.5°F. Thus the analysis shows that the secondary system is not over pressurized and no hot leg boiling occurs in the RCS hot leg. A reduction in the MSSV set point will serve to reduce maximum secondary side temperatures and pressures. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

21. Single Reactor Coolant Pump Locked Rotor (RTSR Section 15.4.4)

This Condition IV event is analyzed under full power conditions assuming the instantaneous seizure of one Reactor Coolant Pump motor. This results in a rapid RCS flow reduction and pressure rise with possible DNB. The reactor is promptly tripped on a low flow signal. The analysis demonstrates that no more than 15% of the rods experience DNB and that the RCS peak pressure remains below that which would cause stresses to exceed the faulted condition stress limits. The secondary system does not reach the MSSV set point during the simulation of this event. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

22. Rupture of a Control Rod Drive Mechanism Housing (RTSR Section 15.4.6)

For this Condition IV event, a rapid reactivity insertion and increase in core power leads to high local fuel and clad temperatures and possible fuel and/or clad damage. Four cases are analyzed: beginning of life, end of life, hot zero power, and hot full power. The analysis shows that the fuel and clad limits discussed in RTSR Section 15.4.6 are not exceeded and that RCS pressure does not exceed the faulted condition stress limits. The MSSV's are not modeled as part of this over pressure analysis and are therefore not required to operate. Thus, the results of this analysis are unaffected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the RTSR remain valid.

23. Steamline Break Mass/Energy Release - Inside/Outside Containment

Various steam line break cases are analyzed for the purposes of generating mass and energy release rates which are then applied to containment response or compartment environmental analyses. Cases are performed assuming various break sizes and initial power levels. For small breaks occurring at high power levels, it is possible that pressurization of the primary and secondary systems may occur. Specifically, if the energy release through the break is less than the decay heat deposition into the RCS, pressurization may occur possibly to the point of safety valve actuation. However, since the relief capacity of the MSSV's is undiminished, there is sufficient capacity to prevent over pressurization of the secondary systems. Raising the MSSV set points will have no impact upon the mass and energy releases previously calculated. Reductions in the MSSV set points will serve to reduce the primary and secondary side temperatures and energy release rates. Thus, the results of the calculated mass and energy releases are not adversely affected by increasing the tolerance on the MSSV's to +/- 3% and the conclusions in the FSAR remain valid.

SET POINT IMPACT

In addition to the impact upon the non-LOCA accident analyses, increasing the MSSV safety valve tolerance to +/- 3% will also impact the core limits and the over-power and over-temperature protection. As seen in Figure 15.1-1 of the VCSNS RTSR, the reactor is protected by the MSSV line. The temperature drop across the steam generator, primary to secondary, is approximately proportional to power. The secondary temperature is approximately constant at the saturation temperature corresponding to the MSSV set point. Therefore the primary temperature cannot rise above the MSSV set point saturation temperature (plus the temperature drop across the steam generator). This temperature limit serves as one of the boundaries on power and temperature in addition to the bounds imposed by the over-power and over-temperature trip set points. By increasing the MSSV set point by 3%, the saturation temperature is increased by approximately 4°F. By decreasing the MSSV set point by 3%, the saturation temperature is decreased by approximately 4°F. Examination of Figure 15.1-1 reveals that movement of the steam generator safety valve line by 4°F will not result in violation of core limits. A reduction in the saturation temperature will in fact reduce the operating space in which the delta-T protection system must provide DNB protection. Thus, the over-power and over-temperature delta-T set points continue to provide protection from core limits.

1

IMAGE EVALUATION
TEST TARGET (MT-3)

1.0 1.1 1.25 1.4 1.6 1.8 2.0 2.2 2.5 2.8 3.2 3.6 4.0

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

LOCA

The following presents the effect of the proposed MSSV set point tolerance increase on the LOCA related analyses.

1. Large Break LOCA (FSAR Chapter 15.4.1)

The licensing basis large break LOCA analysis for VCSNS was performed using the 1981 evaluation model with BASH. The analysis assumed a total core peaking factor (F_q) of 2.45 with uniform 15% steam generator tube plugging. The analysis determined the limiting break size to be a double-ended guillotine with a discharge coefficient equal to 0.4. The peak clad temperature for this case was 2141°F. However due to the fuel load for Cycle 6 being a transition core with Vantage 5 fuel and Standard fuel existing in the core, a transition core penalty of 50°F was assessed. Thus the effective peak clad temperature was determined to be 2191°F.

The large break LOCA analysis does not model the MSSV's. This is because the RCS is quickly depressurized below that of the steam generator secondary pressure and the MSSV's are never challenged. Thus the large break analysis results are not dependent on the performance of the MSSV's. Therefore, the large break LOCA analysis results are not adversely affected by the revised MSSV set point tolerances.

2. Small Break LOCA (FSAR Chapter 15.3.1)

The licensing basis small break LOCA analysis for VCSNS was performed using the NOTRUMP computer code. The analysis assumed a total core peaking factor of (F_q) of 2.50 with 15% steam generator tube plugging. The small break LOCA analysis assumed the plant was operating in Mode 1 at 102% reactor power. The analysis considered break sizes of 2, 3, and 4 inch diameters and determined the limiting break size to be a 3 inch diameter break located in the cold leg. The limiting peak clad temperature was 2095°F.

The small break LOCA analysis requires the MSSV's to remove decay heat from the RCS. Since the small break LOCA assumes loss of offsite power with reactor trip, no credit is taken for operation of the steam generator power operated relief valves or the steam dump system. After reactor trip, the secondary pressure quickly reaches 1225 psia. However, after this initial spike, secondary pressure remains at the first safety lift pressure for the remainder of the transient. Since the tolerance for the first MSSV set point pressure remains unchanged and the maximum pressure obtained is well below the maximum MSSV set point, the small break LOCA analysis is unaffected by the increase in MSSV tolerance to +/- 3% for the four highest set pressures. Therefore, the conclusions in the FSAR remain valid.

3. Hot Leg Switchover to Prevent Potential Boron Precipitation (FSAR Chapter 6.3.2.5)

Post-LOCA hot leg recirculation switchover time is determined for inclusion in emergency procedures to ensure no boron precipitation in the reactor vessel following boiling in the core. This time is dependent on power level, boron concentrations, and water volumes of the RCS, RWST, and accumulators. Since the secondary safety valves affect neither the maximum boron concentrations nor the volumes assumed for the RCS, RWST, and accumulators, there is no effect on the post-LOCA hot leg switchover time.

4. Blowdown Reactor Vessel and Loop Forces (FSAR Chapter 3.9.3)

The blowdown hydraulic loads resulting from a loss of coolant accident are considered in section 3.9.3 of the VCSNS FSAR. Because the maximum loads are generated so quickly, a change in the secondary safety valve set point tolerances would have no effect on the analysis results. Thus, it can be concluded that the consequences of the blowdown reactor vessel and loop forces calculations will not be affected by the revised MSSV set point tolerances.

5. Post-LOCA Long Term Core Cooling; Westinghouse Licensing Position (FSAR Chapter 15.4.1)

The Westinghouse licensing position for satisfying the requirements of 10CFR Part 50 section 50.46 paragraph (b) item (5) "Long Term Cooling" is defined in WCAP-8339. The Westinghouse Evaluation Model commitment is that the reactor will remain shut down indefinitely by borated ECCS water residing in the sump following the postulated LOCA and when SI switchover is accomplished. Since credit for the control rods is not taken for large break LOCA, the borated ECCS water provided by the accumulators and the RWST must have a boron concentration that, when mixed with other water sources, will result in the reactor core remaining subcritical assuming all control rods out.

Sump boron concentration is determined by the accumulation of all potential water sources in the containment, based on each respective source boron concentration. The revised secondary safety valve set point tolerance will not affect the post-LOCA sump boron concentration. It is therefore concluded that there would be no change to the long term cooling capability of the ECCS system as a result of the revised MSSV set point tolerance.

STEAM GENERATOR TUBE RUPTURE

The FSAR analysis for a steam generator tube rupture (SGTR) is performed to evaluate the radiological consequences due to the SGTR event. The major factors that affect the radiological consequences for a SGTR are the amount of radioactivity in the reactor coolant, the amount of reactor coolant transferred to the secondary side of the affected steam generator through the ruptured tube, and the amount of steam released from the steam generator to the atmosphere.

A SGTR results in a decrease in pressurizer pressure due to the loss of reactor coolant inventory. Reactor trip and SI actuation were assumed to occur as a result of low pressure for the VCSNS SGTR analysis. A loss of offsite power was also assumed to occur at the time of reactor trip and thus, the steam dump system was assumed to not be available. The energy transfer from the primary system following reactor and turbine trip causes the secondary side pressure to increase rapidly after reactor trip until the steam generator power operated relief valves (PORV's) and/or safety valves lift to dissipate energy. For the SGTR analysis in the VCSNS FSAR, it is assumed that the secondary pressure is maintained at the lowest secondary safety valve (MSSV) set point following reactor trip. After reactor trip and SI initiation, the RCS pressure was assumed to reach equilibrium at the point where the incoming SI flow rate equals the outgoing break flow rate, and the equilibrium pressure and break flow rate were assumed to persist until 30 minutes after the accident.

Since the equilibrium break flow rate is a function of the primary to secondary pressure differential, a change in the MSSV set point tolerance to +3% will result in the secondary pressure being maintained at a higher pressure during this 30 minute period thereby decreasing the primary to secondary pressure differential. This will result in an decrease to the primary to secondary break flow and thus, a slight decrease in the atmospheric steam release via the ruptured steam generator. Therefore, for a positive increase in set point tolerance (+3%), the FSAR analysis results would remain conservative.

A change in the MSSV set point tolerance to -3% will result in the secondary pressure being maintained at a lower pressure during the 30 minute period thereby increasing the primary to secondary pressure differential. This will result in an increase to the primary to secondary break flow and the atmospheric steam release via the ruptured steam generator.

It is noted that several safety evaluations for plant changes at VCSNS have been previously performed by Westinghouse. The plant changes include changes to secondary operating level, changes to pressurizer operating level, fuel changes, temperature variations associated with the margin broker program, 15% steam generator tube plugging, and increased high head safety injection flow. Sensitivity studies were performed to determine the bounding primary to secondary break flow and atmospheric steam release via the ruptured steam generator for these changes and the increased MSSV set point tolerance. The results of these sensitivity analyses indicate that the primary to secondary break flow would increase but remains less than the reported result due to conservatism in the VCSNS FSAR analysis. The atmospheric steam release via the ruptured steam generator would be increased by approximately 32% over the result reported in the VCSNS FSAR. Finally, it is noted that the reactor coolant activity assumed for the SGTR analysis in the VCSNS FSAR is based on 1% fuel defects and is assumed to be independent of the transient conditions; therefore this assumption would not be affected by the aforementioned changes.

An evaluation incorporating these bounding mass release results was completed to determine the impact on the offsite radiological doses reported in the VCSNS FSAR for the SGTR event. The results of the analysis indicate that the whole body dose reported in the FSAR remains bounding. However, the thyroid dose will increase by 24% over the approximate 0.43 rem reported in the FSAR. Although these results show an increase in the thyroid dose over those presented in the FSAR, this does not constitute an increase in the consequences of the accident. This judgement is based on the NSAC 125 "Guidelines for 10 CFR 50.59 Safety Evaluations" criteria for "increases in consequences," i.e.; the dose increases are small and the total dose is very low, being well within the NRC definition of a "small fraction" of the 10CFR100 exposure guidelines. This "small fraction" is defined as 30 [illegible] thyroid and 2.5 rem whole body which is 10% of the 10CFR[illegible] guideline values of 300 rem thyroid and 250 rem whole body.

GILBERT SAFETY ANALYSES

While the Westinghouse analyses had no problem with the increased set point tolerance for all 15 MSSV's, the Gilbert analyses would allow the increased tolerance on all but the lowest set bank of MSSV's. The lowest set bank of MSSV's are relied upon to be at set point +/- 1% for two reasons. The first is an operational rather than a safety concern. If the lowest set bank of valves were to be set with a - 3 % tolerance, the blow down closure point would be lower than the no load Steam Generator pressure of 1092 psig. The second reason is to maintain a +/- 1% tolerance on set point so as not to affect the capability of the Emergency Feedwater (EFW) system to provide the required water to the Steam Generators.

The EFW flow calculations are based on the set point of the lowest set bank of MSSV's. Westinghouse states that for the worst case transient, the design flow rate of 82.3% of full rated flow @ 100% reactor power will occur for a <u>short</u> period of time. This is because a reactor trip occurs very quickly in the transient. Once a reactor trip occurs, no additional heat (except for decay heat) is added to the system and secondary system pressure and flow rate quickly decrease from that point on. This is exemplified by the fact that secondary steam flow through the MSSV's never goes above 82.3% of full flow even though the reactor was at 100% power upon initiation of the event. Once the pressure transient has turned around, pressure will quickly drop allowing the MSSV's to close and the requirements for EFW come into play. At this time, since little or no heat is being input to the secondary system, pressure will not get above the set pressure of the first bank of MSSV's. Therefore, the EFW flow calculations are based on the set point of the lowest set bank of MSSV's. **Note:** With the increased set point tolerance, it is possible for the second bank of MSSV's to be set slightly below (10 psi) the first bank of MSSV's. This was analyzed by Gilbert and determined to have an insignificant effect on the EFW flow split, i.e; the flow split still meets the design criteria. Since EFW flow capability still meets the design criteria, there is no effect on the capability of the EFW system to mitigate accidents.

CONCLUSIONS

The capability of the MSSV's to mitigate secondary system over-pressure events is maintained with the increased set point tolerance. Therefore the consequences of an event affected by the MSSV's are not increased. The probability of a MSSV malfunction is not affected by this change. No new previously unanalyzed accidents or equipment malfunctions are introduced by this change.

The Tech. Spec. margin of safety remains unchanged for the increase of MSSV set point tolerance from +/- 1% to +/- 3%. The Tech. Spec. requirements for MSSV set point are based upon the requirements of the ASME B&PV Code Section III. Westinghouse safety analyses, in accordance with the ASME B&PV Code, have shown that anticipated transients coincident with an increased MSSV set point tolerance (+3%) will not cause the secondary system pressure allowables (110% design pressure) to be exceeded. The Tech. Spec. margin of safety is that above the ASME limit of 110% design pressure and below ultimate failure. Thus the Tech. Spec. margin of safety for maximum secondary pressure remains unchanged.

The EFW flow calculations are based on the set point of the lowest set bank of MSSV's. Therefore, the set point tolerance for the lowest set bank of MSSV's will not be changed. Since EFW flow capability will remain in compliance with the design criteria, Tech. Spec. margin for EFW flow will remain unchanged.

The new Main Steam Safety Valve set point tolerances are as follows:

MAIN STEAM SAFETY VALVES			MSSV SET POINT	PROPOSED SET POINT TOLERANCE
MAIN STEAM LOOP A	MAIN STEAM LOOP B	MAIN STEAM LOOP C		
XVS-2806 A	XVS-2806 F	XVS-2806 K	1176 PSIG	+/- 1%
XVS-2806 B	XVS-2806 G	XVS-2806 L	1190 PSIG	+/- 3%
XVS-2806 C	XVS-2806 H	XVS-2806 M	1205 PSIG	+/- 3%
XVS-2806 D	XVS-2806 I	XVS-2806 N	1220 PSIG	+/- 3%
XVS-2806 E	XVS-2806 J	XVS-2806 P	1235 PSIG	+/- 3%

10CFR50.59 SAFETY EVALUATION QUESTIONS

		Yes	No
1.	May the proposed activity increase the probability of occurrence of an accident previously evaluated in the FSAR or FPER?	___	X

Basis: The MSSV's provide protection from over-pressurization of the secondary systems and are actuated after an accident is initiated. The accidental depressurization of the Main Steam system events can be initiated by the opening of a MSSV. Increasing the tolerance on these valves does not create a new failure mode or result in a lift set point that would increase the probability of an inadvertent opening of these valves.

		Yes	No
2.	May the proposed activity increase the consequences of an accident previously evaluated in the FSAR or FPER?	___	X

Basis: The capability to mitigate over pressure events remains within acceptable limits (110% of design pressure maximum). MSSV rated flow remains unchanged. The capability of the EFW system to mitigate events remains unchanged because the EFW system continues to meet the design criteria. As previously discussed, DNBR and PCT values affected by the Non-LOCA accident events remain within the limits specified in the licensing basis documentation. It has been demonstrated that the mass/energy releases inside and outside the containment previously documented in the FSAR remain valid. In addition, a review of the SGTR analyses shows that an increase in safety valve set point tolerance will decrease the primary to secondary pressure differential and decrease the break flow rate. The steam release from the ruptured steam generator would decrease slightly, and will have an insignificant change in the offsite doses. However, a decrease in safety valve set point tolerance will increase the primary to secondary pressure differential and increase the break flow rate. As a result, the steam release from the ruptured steam generator would increase

Verification

Type of Verification	Verifier Signature/Date
Independent Review	Sam Skidmore 1-24-91

Approval

Signature/Date
[signature] 1/30/91

slightl causing a small increase in the offsite thyroid doses. This increase in dose under NSAC 125 guidelines is not considered to constitute an increase in the consequences of the accident because the revised doses are still well within the current NRC acceptance criteria as set forth in the Standard Review Plan.

	Yes	No
3. May the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the FSAR or FPER?	___	X

Basis: The probability of a failure of an MSSV is not affected by the increase in set point tolerance to +/- 3%. By maintaining the system upset design pressure, the probability of a malfunction of any other equipment important to safety is not increased.

	Yes	No
4. May the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR or FPER?	___	X

Basis: As previously discussed, the capability to mitigate over pressure events remains within acceptable limits (110% of design pressure maximum). MSSV rated flow remains unchanged. The capability of the EFW system to mitigate events remains unchanged because the design criteria continues to be met. DNBR and PCT values affected by the Non-LOCA and LOCA accident events remain within the limits specified in the licensing basis documentation. It has been demonstrated that the mass/energy releases inside and outside the containment previously documented in the FSAR remain valid. Although it is in reality a failure of equipment, the SGTR event is considered to be an accident, as such, the effects of a change in MSSV set point tolerance on the SGTR analyses was discussed in question 2. The change in MSSV set point does not impact the ability of any other safety system to perform its intended safety function.

Verification

Type of Verification	Verifier Signature/Date
Independent Review	Sam Skidmore / 1-24-91

Approval

Signature/Date
[signature] 1/30/91

		Yes	No
5.	May the proposed activity create the possibility of an accident of a different type than any previously evaluated in the FSAR or FPER?	___	X

Basis: As previously stated, the MSSV's provide over-pressurization protection for the secondary system. The analyses results as presented in the FSAR remain valid and no new failure mechanisms were determined. Thus, the possibility of an accident which is different than any already evaluated in the FSAR and would not be created as a result of increasing the tolerance on the four highest MSSV set pressures to +/- 3%.

		Yes	No
6.	May the proposed activity create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the FSAR or FPER?	___	X

Basis: Allowing a larger MSSV set point tolerance for the four highest set pressures does not result in any conditions being changed which could result in the malfunction of equipment important to safety different from any evaluated in the FSAR/FPER.

		Yes	No
7.	Does the proposed activity reduce the margin of safety as defined in the basis for any technical specification?	___	X

Basis: As indicated in the above evaluation, the conclusions provided in the FSAR remain valid. All acceptance criteria continue to be met. Therefore, there is no reduction in the margin of safety defined in the bases for any Technical Specification.

Verification		Approval
Type of Verification	Verifier Signature/Date	Signature/Date
Independent review	Sam Skidmore / 1-24-91	[signature] 1/30/91