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SCE&G®
A SCANA COMPANY

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Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTN: Mr. Robert E. Martin

Dear Sir / Madam:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
LICENSE AMENDMENT REQUEST - LAR 05-2926
IMPLEMENTATION OF WCAP-14333-P-A, REV. 1, "PROBABILISTIC RISK ANALYSIS OF THE RPS AND ESFAS TEST TIMES AND COMPLETION TIMES"

Pursuant to 10CFR50.90, South Carolina Electric & Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, hereby requests an amendment to the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS).

The proposed changes will revise TS 3/4.3.1, "Reactor Trip System Instrumentation," and TS 3/4.3.2, "Engineered Safety Feature Actuation System Instrumentation," to implement the Allowed Outage Time and Bypass Test Time changes approved by the NRC in WCAP-14333-P-A, Rev. 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," dated October 1998. The proposed changes in this license amendment request are consistent with the NRC approved Technical Specification Task Force (TSTF) Improved Standard Technical Specification Change Traveler TSTF-418, Rev. 2, "RPS and ESFAS Test Times and Completion Times (WCAP-14333)." The changes to Action 16 of TS 3/4.3.2 are justified based on a risk-informed analysis performed in accordance with Regulatory Guides 1.174 and 1.177. These changes will also result in a revision to the Bases for 3/4.3.1 and 3/4.3.2, "Reactor Trip and Engineered Safety Feature Actuation System Instrumentation." The additional changes proposed to Action 8 of TS 3/4.3.1 are consistent with those contained in NUREG-1431, Rev. 3, "Standard Technical Specifications Westinghouse Plants."

Information contained herein provides the No Significant Hazards Determination. Attachment I provides the TS pages marked up with the proposed changes. Attachment II provides the retyped TS pages.

The VCSNS Plant Safety Review Committee and the Nuclear Safety Review Committee have reviewed and approved the proposed changes. SCE&G has notified the State of South Carolina in accordance with 10CFR50.91(b).

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SCE&G requests approval of the proposed amendment within 12 months of submittal in accordance with the NRC goal for review of license amendment requests. Once approved, the amendment shall be implemented within 60 days.

There are no other TS changes in process that will affect or be affected by this change request. There are no significant changes to any FSAR or FPER sections.

If you have any questions or require additional information, please contact Mr. Robert G. Sweet at (803) 345-4080.

I certify under penalty of perjury that the foregoing is true and correct.

11/15/05
Executed on

Jeffrey B. Archie

AJC/JBA/dr

Enclosures:
Evaluation of the proposed changes
Attachment(s): 3
1. Proposed Technical Specification Changes - Mark-up
2. Proposed Technical Specification Changes - Retyped
3. List of Regulatory Commitments

cc: N. O. Lorick
S. A. Byrne
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CER (C-05-2926)
File (813.20)
DMS (RC-05-0143)

Subject: LICENSE AMENDMENT REQUEST - LAR 05-2926
TECHNICAL SPECIFICATIONS 3/4.3.1 and 3/4.3.2 AND ASSOCIATED BASES

1.0 DESCRIPTION

South Carolina Electric & Gas Company (SCE&G) requests an amendment to revise the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) TS 3/4.3.1, "Reactor Trip System Instrumentation," and TS 3/4.3.2, "Engineered Safety Feature Actuation System Instrumentation," to implement the Allowed Outage Time (AOT) and Bypass Test Time changes approved by the NRC in WCAP-14333-P-A, Rev. 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," dated October 1998. The proposed changes in this license amendment request (LAR) are consistent with the NRC approved Technical Specification Task Force (TSTF) Improved Standard Technical Specification Change Traveler TSTF-418, Rev. 2, "RPS and ESFAS Test Times and Completion Times (WCAP-14333)." The proposed changes to Action 16 of TS 3/4.3.2 are justified based on a risk-informed analysis performed in accordance with Regulatory Guides 1.174 and 1.177. These proposed changes will also result in a revision to the Bases for 3/4.3.1 and 3/4.3.2, "Reactor Trip and Engineered Safety Feature Actuation System Instrumentation." The additional changes proposed to Action 8 of TS 3/4.3.1 are consistent with those contained in NUREG-1431, Rev. 3, "Standard Technical Specifications Westinghouse Plants." The AOT acronym and the term "Completion Time," which is utilized in NUREG-1431 are synonymous.

2.0 PROPOSED CHANGES

The following changes to Technical Specifications 3/4.3.1 and 3/4.3.2 are proposed as justified in WCAP-14333-P-A, Rev. 1:

1. The AOT to restore an inoperable RTS or ESFAS analog channel to operable status before it must be placed in the tripped condition is increased from 6 hours to 72 hours;

2. The time allowed for an RTS or ESFAS analog channel to be bypassed for testing other analog channels is increased from 4 to 12 hours;

3. The time allowed for a Reactor Trip Breaker (RTB) to be bypassed for concurrent surveillance testing of the RTB and automatic trip logic is increased from 2 hours to 4 hours; and

4. The AOT to restore an inoperable train of automatic trip logic (TS 3/4.3.1) and automatic actuation logic (TS 3/4.3.2) or actuation relays (TS 3/4.3.2) to operable status before the unit must be shut down, is increased from 6 hours to 24 hours.

The following changes to Action 16 of TS 3/4.3.2 are justified based on a risk-informed analysis performed in accordance with Regulatory Guides 1.174 and 1.177:

1. The AOT to restore an inoperable ESFAS analog channel (for Reactor Building Pressure High 3, Emergency Feedwater Suction Transfer on Low Pressure, and RWST Level Low-Low) to operable status prior to initiating a unit shutdown is increased from 6 hours to 72 hours;

2. The time allowed for an ESFAS analog channel (for Reactor Building Pressure High 3, Emergency Feedwater Suction Transfer on Low Pressure, and RWST Level Low-Low), to be bypassed for testing other analog channels is increased from 4 to 12 hours;

The following changes to Technical Specification 3/4.3.1 are contained in NUREG-1431, Rev. 3:

1. An Action was added to restore an inoperable RTB to operable status in 1 hour,

2. A Note was added that the RTB may be bypassed for 2 hours for maintenance on the undervoltage or shunt trip mechanisms.

Specifically the proposed changes would revise the following:

2.1 TS 3/4.3.1

Action 2 is revised to increase the AOT to restore an inoperable RTS analog channel to operable status before it must be placed in the tripped condition from 6 to 72 hours and the time allowed for an RTS analog channel to be bypassed for testing is increased from 4 hours to 12 hours for Power Range, Neutron Flux - High Setpoint (RTS Function 2. A), Power Range, Neutron Flux - Low Setpoint (RTS Function 2. B), and Power Range, Neutron Flux - High Positive Rate (RTS Function 3).

2.2 TS 3/4.3.1

Action 6 is revised to increase the AOT to restore an inoperable RTS analog channel to operable status before it must be placed in the tripped condition from 6 to 72 hours and the time allowed for an RTS analog channel to be bypassed for testing is increased from 4 hours to 12 hours for Overtemperature ΔT (RTS Function 7), Overpower ΔT (RTS Function 8), Pressurizer Pressure - Low (RTS Function 9), Pressurizer Pressure - High (RTS Function 10), Pressurizer Water Level - High (RTS Function 11), Loss of Flow - Single Loop (RTS Function 12.A), Loss of Flow - Two Loops (RTS Function 12.B), Steam Generator Water Level - Low-Low (RTS Function 13), Steam/Feedwater Flow Mismatch and Low Steam Generator Level (RTS Function 14), Undervoltage- Reactor Coolant Pumps (RTS Function 15), Underfrequency- Reactor Coolant Pumps (RTS Function 16), and Turbine Trip- Low Fluid Oil Pressure (RTS Function 17.A).

2.3 TS 3/4.3.1

Action 8 is revised to increase the time allowed for an RTB to be bypassed from 2 hours to 4 hours for concurrent surveillance testing of the RTB and automatic trip logic.

Action 8 is also revised to include an Action to restore an inoperable breaker to operable status within 1 hour. Currently Action 8 does not include such an Action, and a unit shutdown must be initiated immediately if an RTB is inoperable. A note is also added that an RTB may be bypassed for up to 2 hours for maintenance on the undervoltage or shunt trip mechanisms. These changes are consistent with NUREG-1431, Rev. 3.

2.4 TS 3/4.3.1

Action 12 is revised to increase the AOT to restore an inoperable train of automatic trip logic to operable status before the unit must be shut down from 6 hours to 24 hours for Safety Injection Input from ESF (RTS Function 18) and Automatic Trip Logic (RTS Function 21).

2.5 TS 3/4.3.2

Action 14 is revised to increase the AOT to restore an inoperable train of automatic actuation logic or actuation relays to operable status before the unit must be shut down from 6 hours to 24 hours for Safety Injection Automatic Actuation Logic and Actuation Relays (ESFAS Function 1.b), Reactor Building Spray Automatic Actuation Logic and Actuation Relays (ESFAS Function 2.b), Containment Isolation - Phase A Isolation Automatic Actuation Logic and Actuation Relays (ESFAS Function 3.a.3), and Containment Isolation - Phase B Isolation Automatic Actuation Logic and Actuation Relays (ESFAS Function 3.b.1).

2.6 TS 3/4.3.2

Action 16 is revised to increase the AOT from 6 hours to 72 hours to restore an inoperable ESFAS analog channel to operable status prior to initiating a unit shutdown, and the time allowed for an ESFAS analog channel to be bypassed for testing is increased from 4 hours to 12 hours for Reactor Building Pressure High 3 (ESFAS Functions 2.c and 3.b.2), Emergency Feedwater Suction Transfer on Low Pressure (ESFAS Function 6.h), and RWST Level Low-Low (ESFAS Function 8.a).

2.7 TS 3/4.3.2

Action 21 is revised to increase the AOT to restore an inoperable train of automatic actuation logic or actuation relays to operable status before the unit must be shut down from 6 hours to 24 hours for Steam Line Isolation Automatic Actuation Logic and Actuation Relays (ESFAS Function 4.b), Emergency Feedwater Automatic Actuation Logic and Actuation Relays (ESFAS Function 6.b), and Automatic Switchover to Containment Sump Automatic Actuation Logic and Actuation Relays (ESFAS Function 8.b). Action 21 is also revised from "restore the inoperable channels", to "restore the inoperable channel", since there are only two trains of automatic actuation logic, and this Action only address one inoperable train of actuation logic.

2.8 TS 3/4.3.2

Action 24 is revised to increase the AOT to restore an inoperable ESFAS analog channel to operable status before it must be placed in the tripped condition is from 6 to 72 hours and the time allowed for an ESFAS analog channel to be bypassed for testing is increased from 4 hours to 12 hours for Safety Injection on Reactor Building Pressure - High 1 (ESFAS Function 1.c), Safety Injection on Pressurizer Pressure - Low (ESFAS Function 1.d), Safety Injection on Differential Pressure Between Steam Lines - High (ESFAS Function 1.e), Safety Injection on Steam Line Pressure - Low (ESFAS Function 1.f), Steam Line Isolation on Reactor Building Pressure - High 2 (ESFAS Function 4.c), Steam Line Isolation on Steam Flow in Two Steam Lines - High and Coincident with Tavg - Low-Low (ESFAS Function 4.d), Steam Line Isolation on Steam Line Pressure - Low (ESFAS Function 4.e), Turbine Trip and Feedwater Isolation on Steam Generator Water Level - High-High (ESFAS Function 5.a), Emergency Feedwater on Steam Generator Water Level - Low-Low Start Motor Driven Pumps (ESFAS Function 6.c.i), and Emergency Feedwater on Steam Generator Water Level - Low-Low Start Turbine Driven Pump (ESFAS Function 6.c.ii).

2.9 TS 3/4.3.2

Action 25 is revised to increase the AOT to restore an inoperable train of automatic actuation logic or actuation relays to operable status before the unit must be shut down from 6 hours to 24 hours for Turbine Trip and Feedwater Isolation Automatic Actuation Logic and Actuation Relays (ESFAS Function 5.b), Action 25 is also revised from "restore the inoperable channels", to "restore the inoperable channel", since there are only two trains of automatic actuation logic, and this Action only address one inoperable train of actuation logic.

2.10 Bases 3/4.3.1 and 3/4.3.2

The Bases are being updated to include a reference to WCAP-14333-P-A, Rev. 1, which justified the changes for the increased AOTs and bypass test time, and a reference to the risk-informed justification for Reactor Building Pressure High 3 (ESFAS Functions 2.c and 3.b.), Emergency Feedwater Suction Transfer on Low Pressure (ESFAS Function 6.h), and RWST Level Low-Low (ESFAS Function 8.a) performed in accordance with Regulatory Guides 1.174 and 1.177.

3.0 BACKGROUND

The Westinghouse Owners Group (WOG) prepared a topical report that justified the relaxation of RTS and ESFAS bypass test times and AOTs for the protection system instrumentation. The relaxations were justified by an analysis of the protection system availability and the impact of that availability change on the overall plant risk. The original justification was identified by the acronym TOP (Technical Specification Optimization Program) as documented in the WCAP-10271-P-A series of reports. Those changes were implemented at VCSNS by License Amendment No. 101 to the VCSNS Operating License, and the Correction Letter for License

Amendment No. 101, References 1 and 2, respectively. When reviewing risk metric results, VCSNS current licensing basis is that of a "TOP" plant. License Amendment 167 revised Action 16 of TS 3/4.3.2 for Reactor Building Pressure High 3 (ESFAS Functions 2.c and 3.b.2), Emergency Feedwater Suction Transfer on Low Pressure (ESFAS Function 6.h), and RWST Level Low-Low (ESFAS Function 8.a) to add an AOT of 6 hours to restore an inoperable channel of these ESFAS Functions to operable status prior to initiating a unit shutdown. The proposed changes addressed these energize to actuate channels and the single failure of a loss of power in the opposite train of instrumentation while a channel is bypassed. The VCSNS instrumentation power supplies to these energize to actuate channels are different than similar functions evaluated in WCAP-14333-P-A, Rev. 1; therefore, a separate risk-informed justification in accordance with Regulatory Guides 1.174 and 1.177 was performed for these channels, which will be discussed later in this LAR.

Fault tree models of the protection system instrumentation were used to calculate the unavailability impact due to changes to test and maintenance time allowances. The changes in RTS and ESFAS unavailability were then used in a risk model to determine changes in core damage frequency (CDF) and large early release frequency (LERF) as the test and maintenance time allowances were relaxed. Differences in the analysis methods between the TOP, WCAP-10271-P-A series of reports (hereafter referred to as WCAP-10271) and WCAP-14333 are discussed in Section 7.1 of WCAP-14333-P-A, Rev. 1.

The approach used in WCAP-14333-P-A, Rev. 1, is consistent with the approach established in the TOP program. This includes the fault tree models, signals, component reliability database, and most of the test and maintenance assumptions. The methodology in WCAP-14333-P-A, Rev. 1, used a representative set of RTS and ESFAS signals to determine the impact of the proposed changes on signal unavailability, and used a representative PRA model to determine the impact on CDF and LERF. In comparison to WCAP-10271, WCAP-14333-P-A, Rev. 1, used a different common cause failure modeling approach for analog channels and included more realistic assumptions related to the component unavailability due to maintenance activities based on a survey of WOG plants. Operator actions to either manually trip the reactor or initiate safety injection are also modeled in WCAP-14333-P-A, Rev. 1. In addition, credit for emergency feedwater pump start from the Anticipated Transient Without a Scram (ATWS) mitigating system actuation circuitry (AMSAC) was taken. More discussion of these differences is contained in Sections 7 and 8 of WCAP-14333-P-A, Rev. 1. The relaxations that are justified in WCAP-14333-P-A, Rev.1 are summarized below:

Summary of WCAP-14333-P-A, Rev. 1, RTS and ESFAS Completion Time and Bypass Test Time Changes - Solid State Protection System		
Component	**Completion Time**	**Bypass Test Time**
Analog channels	6+6 hours to 72+6 hours	4 hours to 12 hours
Logic train	6+6 hours to 24+6 hours	no relaxation*
Actuation relays	6+6 hours to 24+6 hours	no relaxation*
*no relaxation beyond TOP (WCAP-10271 and Supplements)		

WCAP-14333 was submitted for NRC review and approval by WOG letter OG-95-51 dated June 20, 1995. The NRC issued a Safety Evaluation (SE) on July 15, 1998, approving WCAP-14333. These improvements will allow additional time to perform maintenance and test activities, enhance safety, provide additional operational flexibility, and reduce the potential for forced outages related to compliance with the RTS and ESFAS instrumentation Technical Specifications. Industry information has shown that trips have occurred during instrumentation test and maintenance activities, indicating that these activities should be completed with caution and sufficient time should be available to complete these activities in an orderly and effective manner.

Southern Nuclear Operating Company (SNC) submitted a LAR on October 13, 1999, for the Vogtle Units 1 and 2 to implement the relaxations that were generically approved in WCAP-14333, Rev. 1. In response to an NRC request for additional information (RAI) on the SNC LAR, incremental conditional large early release probability (ICLERP) values were calculated that are generically applicable to all WOG plants. Reference 4 identifies the SNC LAR correspondence. Amendments 116 and 94 were issued for Vogtle Units 1 and 2 approving the changes justified in WCAP-14333-P-A, Rev 1 (TAC Numbers MC 4977 and MC 4978).

4.0 TECHNICAL ANALYSIS

A survey was provided to all WOG members to determine their needs with respect to instrumentation test times, maintenance times, and maintenance frequencies, in addition to information regarding plant operation such as reactor trip and spurious safety injection events. The Technical Specification changes evaluated in WCAP-14333 were identified from the survey information. The probabilistic risk analysis, benefits of the program and conclusions, and the relationship of the Technical Specification changes to the analyses are discussed in WCAP-14333-P-A, Rev. 1.

In order to model the Allowed Outage Times in the fault trees to determine the impact of the changes on signal unavailabilities, several parameters were specified for component test and maintenance unavailabilities. These are the test frequencies and durations discussed in Section 5.1 of WCAP-14333-P-A, Rev. 1, and the maintenance frequencies and durations discussed in Section 5.2 of WCAP-14333-P-A, Rev. 1.

The changes considered in this analysis were evaluated consistent with the three-tiered approach currently defined in Regulatory Guide 1.177. The first tier addresses PRA insights and includes the risk analyses and sensitivity analyses to support the Allowed Outage Time and bypass test time changes. The second tier addresses avoidance of risk-significant plant configurations. The third tier addresses risk-informed plant configuration control and management.

The following Tier 1, Tier 2, and Tier 3 discussions are associated with WCAP-14333-P-A, Rev. 1.

Tier 1: PRA Capability and Insights

WCAP-14333-P-A, Rev. 1, originally provided only the impact of the requested changes on core damage frequency (ΔCDF) for a two-out-of-four (2/4) and two-out-of-three (2/3) actuation logic. In the responses to NRC RAI Questions 11 and 13 in WOG letter OG-96-110 (Reference 5), the WOG provided the incremental conditional core damage probability (ICCDP) for various components in maintenance for the proposed changes, and the change in large early release frequency (ΔLERF) for 2/4 and 2/3 actuation logic for the proposed changes. Additionally, in response to an NRC RAI on SNC's LAR to implement these changes for the Vogtle Units 1 and 2 (Reference 4), generic ICLERP values for various components in maintenance were provided.

The impact of the proposed changes on CDF and LERF is provided in TSTF-418, Rev. 2, Table 1.3 (which is the same information that is contained in Table 8.4 of WCAP-14333-P-A, Rev. 1) and Table 1.4 (which is the same information that is contained in Table Q13.1 in the response to RAI Question 13 in OG-96-110), respectively. The CDF and LERF values are provided for pre-TOP, TOP, and the WCAP-14333-P-A, Rev. 1, proposed changes. The ΔCDF and ΔLERF values are also provided and referenced to the pre-TOP and TOP conditions. The results of a sensitivity analysis are also provided that credits a 0.5/year reduction in reactor trip frequency due to fewer analog channel operational tests (ACOT) (trip reduction originally postulated for the WCAP-10271 ACOT interval increase from monthly to quarterly). The ΔCDF and ΔLERF values are provided for both a 2/4 and 2/3 logic. The ICCDP and ICLERP values are provided on Table 1.5 of TSTF-418, Rev. 2 (the ICCDP values from Table Q11.1 in the response to RAI Question 11 in OG-96-110, Reference 5 and the ICLERP values in SNC's RAI response in Reference 4). The ICCDP and ICLERP values are provided only for 2/3 logic, however the results bound the 2/4 logic.

Combined Risk Metric Results
(From Tables 1.3, 1.4, and 1.5 in TSTF-418, Rev. 2)

Risk Metric	Acceptance Criterion	Change from WCAP-10271 to WCAP-14333-P-A, Rev. 1	
ΔCDF per year	< 1E-06	2/4 logic 3.5E-07	2/3 logic 6.1E-07
ICCDP	< 5E-07	Ranges from 4.4E-07 (logic train in maintenance) to 5.5E-10 (SG level channel in test)*	
ΔLERF per year	< 1E-07	2/4 logic 2.0E-08	2/3 logic 2.2E-08
ICLERP	< 5E-08	Ranges from 3.0E-08 (logic train in maintenance) to 1.1E-11 (SG level channel in test)*	

*The ICCDP and ICLERP values are provided only for 2/3 logic, however the results bound the 2/4 logic.

The ICCDP and ICLERP values are dependent on the particular component in test or maintenance. The acceptance criteria as defined in Regulatory Guide 1.177 for these incremental risk metrics are satisfied. The ΔCDF and ΔLERF values are from the current licensing basis (WCAP-10271) to the proposed state (WCAP-14333-P-A, Rev. 1), and do not credit the 0.5/year reduction in reactor trip frequency.

Tier 2: Avoidance of Risk-Significant Plant Configurations

Tier 2 requires an examination of the need to impose additional restrictions when operating in the proposed AOTs in order to avoid risk-significant equipment outage configurations.

In support of Tier 2 limitations, analyses were completed in response to an NRC RAI on WCAP-14333. RAI #18 asked for other risk significant systems or components for the proposed test or maintenance plant configuration. Analyses were completed in support of the response to RAI Question 18 in Reference 5, that determined the system importance for plant configurations with no ongoing test and maintenance activities (all components available), and for plant configurations with ongoing test or maintenance activities individually on the analog channels, logic cabinets, master relays, and slave relays. With test or maintenance activities in progress, it was assumed that the corresponding component or train will be unavailable. The system importance for these configurations is provided in Table Q18.1 in the response to RAI Question 18 on WCAP-14333 in OG-96-110, Reference 5. The importance values were compared between the cases with individual components unavailable and all components available. The following was concluded:

- The importance rankings of risk significant systems do not change appreciably for the configurations with an analog channel, master relay, or slave relay out of service, or unavailable, compared to the configuration with no ongoing test or maintenance activities.

- A relatively significant change in the importance rankings of risk significant systems occurs when a logic cabinet is out of service, or unavailable, compared to the configuration with no ongoing test or maintenance activities.

RAI Question 11 on WCAP-14333 requested CDFs for the various test and maintenance configurations that the plant would enter for the subject AOTs extensions. Conditional core damage frequencies and core damage probabilities were calculated for each of the possible test and maintenance configurations. This information is provided in Table Q11.1 in the response to RAI Question 11 on WCAP-14333 in OG-96-110, Reference 5. It was concluded from the information contained in Table Q11.1 that the only configuration that significantly impacts CDF was with a logic cabinet unavailable.

Based on the information provided in Tables Q11.1 and Q18.1 in the responses to RAI Questions 11 and 18 on WCAP-14333 in OG-96-110, Reference 5, it was concluded that the only plant configuration with an appreciable impact on CDF or a significant impact on the relative importance of other systems is the configuration with one logic cabinet unavailable.

Therefore, Tier 2 limitations are only appropriate when a logic cabinet is out of service. There are no Tier 2 limitations when a slave relay, master relay, or analog channel is out of service.

Consistent with the SE requirements to include Tier 2 insights into the decision making process before taking equipment out of service, restrictions on concurrent removal of certain equipment when a logic cabinet is unavailable are as follows:

- To preserve ATWS mitigation capability, activities that degrade the availability of the emergency feedwater system, RCS pressure relief system (pressurizer PORVs and safety valves), AMSAC, or turbine trip should not be scheduled when a logic cabinet is unavailable.

- To preserve LOCA mitigation capability, one complete ECCS train that can be actuated automatically must be maintained.

- To preserve reactor trip and safeguards actuation capability, activities that cause master relays or slave relays in the available train and activities that cause analog channels to be unavailable should not be scheduled when a logic cabinet is unavailable.

- Activities on electrical systems (e.g., AC and DC power) and cooling systems (e.g., service water and component cooling water) that support the systems or functions listed in the first three bullets should not be scheduled when a logic cabinet is unavailable. That is, one complete train of a function that supports a complete train of a function noted above must be available.

Tier 3: Risk-Informed Configuration Risk Management

Tier 3 requires a proceduralized process to assess the risk associated with both planned and unplanned work activities. The objective of the third tier is to ensure that the risk impact of out-of-service equipment is evaluated prior to performing any maintenance activity. As stated in Section 2.3 of Regulatory Guide 1.177, "a viable program would be one that is able to uncover risk-significant plant equipment outage configurations in a timely manner during normal plant operation." The third-tier requirement is an extension of the second-tier requirement, but addresses the limitation of not being able to identify all possible risk-significant plant configurations in the second-tier evaluation. Programs and procedures are in place at VCSNS to address this objective.

The risk impact associated with the performance of maintenance and testing activities is evaluated in accordance with a VCSNS Operations Administrative Procedure. This procedure ensures that configuration risk is assessed using a PRA-based model and managed prior to initiating any maintenance activity consistent with the requirements of 10CFR50.65(a)(4). This procedure also ensures that risk is reassessed if an emergent condition results in a plant configuration that has not been previously assessed. Risk thresholds are established to ensure that the average baseline risk is maintained within an acceptable band. When administrative limits are exceeded, increasing levels of management approval are required prior to initiating the work.

WCAP-14333-P-A, Rev. 1, SE Conditions

The NRC approval of WCAP-14333 was subject to the following conditions requiring plant-specific information:

1. Confirm the applicability of the WCAP-14333 analyses to the plant.

2. Address the Tier 2 and 3 analyses including the Configuration Risk Management Program (CRMP) insights which confirm that these insights are incorporated into the decision making process before taking equipment out of service.

WCAP-14333-P-A, Rev. 1, SE Condition 1, Topical Report Applicability Determination

In order to address SE Condition 1, Westinghouse issued implementation guidelines for licensees to use to confirm the analyses are applicable to their plant. The following tables have been completed to confirm the applicability of WCAP-14333-P-A, Rev. 1, to VCSNS. If the plant specific parameters are consistent with the analysis assumptions of WCAP-14333-P-A, Rev. 1, identified in these tables, then the WCAP-14333-P-A, Rev. 1, analysis and results are applicable to the plant.

Table 1: WCAP-14333-P-A, Rev. 1, Implementation Guidelines: Applicability of the Analysis - General Parameters		
Parameter	**WCAP-14333-P-A, Rev. 1, Analysis Assumptions**	**Plant Specific Parameter**
Logic Cabinet Type (1)	Relay and SSPS	SSPS
Component Test Intervals (2)		
• Analog channels	3 months	3 months
• Logic cabinets (SSPS)	2 months	2 months
• Logic cabinets (Relay)	1 month	NA
• Master Relays (SSPS)	2 months	2 months
• Master Relays (Relay)	1 month	NA
• Slave Relays	3 months	3 months
• Reactor trip breakers	2 months	2 months
Analog Channel Calibrations (3)		
• Done at-power	Yes	Yes (Partial)
• Interval	18 months	Equal to or greater than
Typical At-Power Maintenance Intervals (4)		
• Analog channels	24 months	Equal to or greater than
• Logic cabinets (SSPS)	18 months	Equal to or greater than
• Logic cabinets (Relay)	12 months	NA
• Master relays (SSPS)	Infrequent (5)	Infrequent
• Master relays (Relay)	Infrequent (5)	NA
• Slave relays	Infrequent (5)	Infrequent
• Reactor trip breakers	12 months	Equal to or greater than
AMSAC (6)	Credited for EFW pump start	Yes
Total Transient Event Frequency (7)	3.6/yr	1.02/yr
ATWS Contribution to CDF (current PRA model) (8)	8.4E-06/yr	2.7E-07/yr
Total CDF from Internal Events (current PRA model) (9)	5.8E-05/yr	6.08E-5 (see Explanatory Note A)
Total CDF from Internal Events (IPE) (10)	Not Applicable	2.0E-4

Notes for Table 1

1. Indicate the type of logic cabinet: SSPS or Relay (both are included in WCAP-14333-P-A, Rev. 1).
2. Fill in the applicable test intervals. If the test intervals are equal to or greater than those used in WCAP-14333-P-A, Rev. 1, the analysis is applicable.
3. Indicate if channel calibrations are done at-power and, if so, fill in the interval. If channel calibrations are not done at-power or if the calibration interval is equal to or greater than that used in WCAP-14333-P-A, Rev. 1, the analysis is applicable.
4. Fill in the applicable typical maintenance intervals or fill in "equal to or greater than" or "less than." If maintenance intervals are equal to or greater than those used in WCAP-14333-P-A, Rev. 1, the analysis is applicable.
5. Only corrective maintenance is done on the master and slave relays. The maintenance interval on typical relays is relatively long, that is, experience has shown that they do not typically completely fail. Failure of slave relays usually involves failure of individual contacts. Fill in "infrequent" if this is consistent with the plant experience. If not, fill in the typical maintenance interval. If "infrequent" slave relay failures are the norm, then the WCAP-14333-P-A, Rev. 1, analysis is applicable.
6. Indicate if AMSAC will initiate EFW pump start. If yes, then the WCAP-14333-P-A, Rev. 1, analysis is applicable.
7. Include the total frequency for initiators requiring a reactor trip signal to be generated for event mitigation. This is required to assess the importance of ATWS events to CDF. Do not include events initiated by a reactor trip.
8. Fill in the ATWS contribution to CDF (from at-power, internal events). This is required to determine if the ATWS event is a large contributor to CDF.
9. Fill in the total CDF from internal events (including internal flooding) for the most recent PRA model update. This is required for comparison to the NRC's risk-informed CDF acceptance guidelines.
10. Fill in the total CDF from internal events from the IPE model (submitted to the NRC in response to Generic Letter 88-20). If this value differs from the most recent PRA model update CDF, provide a concise list of reasons, in bulletized form, describing the differences between the models that account for the change in CDF.

Explanatory Note A: Note that the total CDF from internal events in Table 1 for VCSNS is slightly greater than the WCAP-14333 analysis value. The difference is acceptable for the following two reasons:

- The absolute CDF value is not important to the acceptability of the results. The important parameter was the change in CDF.
- The VCSNS PRA model results are not dominated by signal failures. Signal failures are small contributors to CDF.

Table 2: WCAP-14333-P-A, Rev. 1, Implementation Guidelines: Applicability of Analysis Reactor Trip Actuation Signals		
Event	**WCAP-14333-P-A, Rev. 1, Analysis Assumption**	**Plant Specific Parameter (1)**
Large LOCA	Not Required	Agree
Medium LOCA	Not Required	Agree
Small LOCA	Nondiverse (2) w/OA (3)	Agree
Steam Generator Tube Rupture	Nondiverse w/OA	Agree
Interfacing System LOCA	Not Required	Agree
Reactor Vessel Rupture	Not Required	Agree
Secondary Side Breaks	Nondiverse w/OA	Agree
Transient Events, such as: - Positive Reactivity Insertion - Loss of Reactor Coolant Flow - Total or Partial Loss of Main Feedwater - Loss of Condenser - Turbine Trip - Loss of DC Bus - Loss of Vital AC Bus - Loss of Instrument Air - Spurious Safety Injection - Inadvertent Opening of a Steam Valve	Diverse (4) w/OA	Agree
Reactor Trip	Generated by RPS	Agree
Loss of Offsite Power	Not Required by RPS	Agree
Station Blackout	Not Required by RPS	Agree
Loss of Service Water or Component Cooling Water	Nondiverse w/OA	Agree

Notes for Table 2

1. Fill in "agree" if the plant design and operation is consistent with this analysis, that is, the noted reactor trip signals are available at a minimum. If not, explain the difference. If "agree" is listed for each event, then the WCAP 14333-P-A, Rev. 1, analysis is applicable.
2. Nondiverse means that (at least) one signal will be generated to initiate a reactor trip for the event.
3. OA indicates that an operator could take action to initiate a reactor trip for the event, that is, there is sufficient time for operator action and procedures are in place that will instruct the operator to take the action.
4. Diverse means that (at least) two signals will be generated to initiate a reactor trip for the event.

Table 3: WCAP-14333-P-A, Rev. 1, Implementation Guidelines: Applicability of Analysis Engineered Safety Features Actuation Signals			
Safety Function	**Event**	**WCAP-14333-P-A, Rev. 1, Analysis Assumption**	**Plant Specific Parameter (1)**
Safety Injection	Large LOCA	Nondiverse (2)	Agree
	Medium LOCA	Nondiverse, OA (3) by SI switch on main control board	Agree
	Small LOCA	Nondiverse, OA by SI switch on main control board, OA of individual components	Agree
	Interfacing Systems LOCA	Nondiverse, OA by SI switch on main control board, OA of individual components	Agree
	SG Tube Rupture	Nondiverse, OA by SI switch on main control board, OA of individual components	Agree
	Secondary Side Breaks	Nondiverse, OA by SI switch on main control board, OA of individual components	Agree
Emergency Feedwater Pump Start	Events generating SI signal Transient events	Pump actuation on SI signal Nondiverse, AMSAC, operator action	Agree
Main Feedwater Isolation	Secondary Side Breaks	Nondiverse	Agree
Steamline Isolation	Secondary Side Breaks	Nondiverse	Agree
Containment Spray Actuation	All events	Nondiverse signal	Agree
Containment Isolation	All events	From SI signal	Agree
Containment Cooling	All events	From SI signal	Agree

Notes for Table 3

1. Fill in "agree" if the plant design and operation is consistent with this analysis, that is, the noted Engineered Safety Features Actuation Signals are available at a minimum. If not, explain the difference. If "agree" is listed for each event, then the WCAP 14333-P-A, Rev. 1, analysis is applicable.
2. Nondiverse means that (at least) one signal will be generated to initiate the engineered safety feature noted for the event.
3. OA indicates that an operator could take action to initiate the engineered safety feature for the event, that is, there is sufficient time for operator action and procedures are in place that will instruct the operator to take the action.

Based on the completion of these tables, it is concluded that the WCAP-14333-P-A, Rev. 1, analyses are applicable to VCSNS.

WCAP-14333-P-A, Rev. 1, SE Condition 2

SE Condition 2 is addressed by the Tier 2 and Tier 3 discussions above.

Plant Specific Functions not Generically Evaluated

A risk-informed analysis was performed in accordance with Regulatory Guide 1.174, using the VCSNS PRA model to assess the impact of the changes on CDF and LERF of increasing the AOT from 6 hours to 72 hours to restore an inoperable ESFAS analog channel to operable status prior to initiating a unit shutdown, and increasing the time allowed for an ESFAS analog channel to be bypassed for testing from 4 hours to 12 hours for Reactor Building Pressure High 3 (ESFAS Functions 2.c and 3.b.2), Emergency Feedwater Suction Transfer on Low Pressure (ESFAS Function 6.h), and RWST Level Low-Low (ESFAS Function 8.a). In accordance with Regulatory Guide 1.177 the ICCDP and ICLERP values were also determined. The VCSNS instrumentation power supplies to these energize to actuate channels are different than similar functions evaluated in WCAP-14333-P-A, Rev.1, therefore a separate risk-informed justification in accordance with Regulatory Guides 1.174 and 1.177 was performed for these channels.

Tier 1: PRA Capability and Insights

The risk results of the above changes for Reactor Building Pressure High 3 (ESFAS Function 2.c), Emergency Feedwater Suction Transfer on Low Pressure (ESFAS Function 6.h), and RWST Level Low-Low (ESFAS Function 8.a) are provided in Table 4 below.

Table 4: Risk Metrics for the Technical Specification Changes to Reactor Building Pressure High 3 (ESFAS Function 2.c), Emergency Feedwater Suction Transfer on Low Pressure (ESFAS Function 6.h), and RWST Level Low-Low (ESFAS Function 8.a)

Signal	ΔCDF	ΔLERF	ICCDP	ICLERP
Reactor Building Pressure High 3 (ESFAS Function 2.c)	0.0E-00/yr	0.0E-00/yr	0.0E-00	0.0E-00
Emergency Feedwater Suction Transfer on Low Pressure (ESFAS Function 6.h)	0.0E-00/yr	0.0E-00/yr	0.0E-00	0.0E-00
RWST Level Low-Low (ESFAS Function 8.a)	3.8E-08/yr	4.0E-10/yr	1.8E-08	1.4E-10
Combined impact	3.8E-08/yr	4.0E-10/yr	--	--

Note that all the risk metrics in Table 1 meet the acceptance criteria provided in RG 1.174 and RG 1.177. These guidelines are:

- ΔCDF < 1E-06/yr
- ΔLERF < 1E-07/yr
- ICCDP < 5E-07
- ICLERP< 5E-08

Reactor Building Pressure High 3 (ESFAS Function 3.b.2) was not evaluated quantitatively since it is not explicitly modeled in the VCSNS PRA model. The Phase B containment isolation signal (Reactor Building Pressure High 3 [ESFAS Function 3.b.2]) is used to isolate component cooling water (CCW) to the containment, CCW to the thermal barrier heat exchangers, and chemical feed from the emergency feedwater system to the steam generators. Each of these systems is a closed system inside containment, that is, it is closed to the containment atmosphere with piping, or a similar component, forming the barrier. With such systems, a pathway from the containment atmosphere to the outside environment requires a pipe break inside containment in addition to a failure of the containment isolation valve (CIV) outside the containment to close and/or remain closed. Typically, such penetrations are screened out of the containment isolation analyses due to the very low probability of the required pipe break. Given this, CIVs associated with these penetrations are not included in the PRA model. Since this type of penetration is screened from the analysis, i.e., it is considered to contribute negligibly to releases, the Phase B containment isolation signal will also be unimportant to containment isolation success. Therefore, the changes for the Phase B containment isolation signal will have a negligible impact on the LERF.

Tier 2: Avoidance of Risk-Significant Plant Configurations

See the previous discussion for WCAP-14333-P-A, Rev. 1.

Tier 3: Risk-Informed Configuration Risk Management

See the previous discussion for WCAP-14333-P-A, Rev. 1.

Impact on Defense-in-Depth

The proposed changes consider the defense-in-depth principle consisting of a number of elements. These elements and the impact of the proposed changes on each are discussed below:

- A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved.

The proposed changes to the AOTs and bypass test times have only a very small calculated impact on CDF and LERF as previously discussed. These changes do not degrade core damage prevention and compensate with improved containment integrity, nor do these changes degrade containment integrity and compensate with improved core damage prevention. The balance between prevention of core damage and prevention of containment failure is maintained. Consequence mitigation remains unaffected by the proposed changes. Furthermore, no new accident or transients are introduced with the proposed changes and the likelihood of accidents or transients is not impacted. Conversely, increased AOTs and bypass test times have the potential to lead to a reduction in the likelihood of transients or accidents caused by maintenance or tests. The additional time to complete these activities provides an atmosphere more conducive to successfully completing repair and test activities without

inducing a plant transient and reducing system re-alignment and restoration errors. These remain unquantified benefits of the proposed changes.

- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.

The plant design will not be changed with these proposed changes. All safety systems, including the reactor protection system signals, will still function in the same manner with the same reliability, and there will be no additional reliance on additional systems, procedures, or operator actions. The calculated risk increase for the proposed changes is very small and additional control processes are not required to compensate for any risk increase.

- System redundancy, independence, and diversity are preserved commensurate with the expected frequency and consequences of challenges to the system, and uncertainties.

There is no impact on the redundancy, independence, or diversity of the reactor protection system or on the ability of the plant to respond to events with diverse systems. The reactor protection system has both diversity and redundancy built into the design and is backed up by operator actions. This will not be impacted by the proposed changes. These signals are highly reliable and will remain so after the proposed changes are implemented.

- Defenses against potential common cause failures are maintained, and the potential for introduction of new common cause failure mechanisms is assessed.

Defenses against common cause failures are maintained. The proposed changes are not sufficiently long to expect new common cause failure mechanisms to arise. In addition, the operating environment for these components remains the same. Again, new common cause failures modes are not expected. In addition, backup systems are not impacted by these changes and no new common cause links between the primary and backup systems are introduced. Therefore, no new potential common cause failure mechanisms have been introduced by the proposed changes.

- Independence of physical barriers is not degraded.

The barriers protecting the public and the independence of these barriers are maintained. With the proposed changes it is not expected that multiple systems will be out of service simultaneously that could lead to degradation of these barriers and an increase in risk to the public to occur. In addition, the extended AOTs and bypass test times do not provide a mechanism that degrades the independence of the fuel cladding, reactor coolant system, and containment barriers.

- Defenses against human errors are maintained.

No new operator actions related to the proposed changes are required to maintain plant safety. No additional operating, maintenance, or test procedures have been introduced or modified due to these changes. The increases in AOTs and bypass test times provide additional time to

complete troubleshooting and test and repair activities which will result in improved operator and maintenance personnel performance resulting in reduced system re-alignment and restoration errors.

Impact on Safety Margins

The safety analysis acceptance criteria stated in the Final Safety Analysis Report (FSAR) is not impacted by these proposed changes. Redundancy and diversity of the reactor protection system will be maintained. The proposed changes will not allow plant operation in a configuration outside the design basis. All reactor protection system signals credited in the safety analyses will remain the same.

VCSNS PRA Quality

The VCSNS PRA model has been the subject of a peer review by the Westinghouse Owners Group. All A level Facts and Observations (F&Os) have been addressed and all but two of the B level F&Os have been addressed. The two B level F&Os that remain to be addressed are provided below. As discussed below, these B level F&Os will have no impact on the applicability or implementation of the changes justified in WCAP-14333-P-A, Rev. 1, to VCSNS since the analysis supporting this WCAP does not rely on plant specific PRA models. Additionally, these B level F&Os will not impact the changes to the energize to actuate channels as discussed below.

B Level F&Os To Be Addressed

HR-06: It is not clear that the full plant level perspective of the symptoms and plant conditions that may influence the time available to perform Type C actions have been adequately taken into account. For example for sequences involving operator actions after a loss of CCW or loss of SW initiating events, it was not evident that the interactions and complexities associated with the plant being in multiple procedures at the same time was taken into account. The Human Reliability Analysis (HRA) evaluation of these actions makes reference to the loss of CCW procedure but does not explicitly address the additional procedures such as emergency operating procedures to cope with loss of CCW to a charging pump and CVCS heat exchangers, etc. that the operators will be involved with during the scenario. Hence when the time window is compared with the time needed to complete a given action, the time needed to address concurrent activities is not explicitly considered.

This issue also relates to the treatment of human action dependencies in the following respect. The Human Error Probability (HEP) values including the time window analysis is done for sequences independent of the underlying cutsets. Some of the cutsets involve concurrent human actions whose time to complete will be competing with those of a given action. Hence for these cases, the time windows should be further adjusted.

Assessment of Impact

The subject of this F&O concerns accounting for complexities associated with Type C (post initiator) operator actions. The complexities are associated with being in multiple procedures simultaneously and dependencies between human actions. In the WCAP analysis, operator actions are credited for manually tripping the reactor, initiating safety injection, and for starting or aligning safety systems that failed to start or align automatically. Since a generic analysis was done in WCAP-14333-P-A, Rev. 1, to demonstrate the acceptability of the proposed changes, this F&O has no impact on the signals addressed generically in the WCAP.

With regard to the Reactor Building Pressure High 3 (ESFAS Functions 2.c and 3.b.2), Emergency Feedwater Suction Transfer on Low Pressure (ESFAS Function 6.h), and RWST Level Low-Low (ESFAS Function 8.a) channels, the following are concluded:

Reactor Building Pressure High 3 (ESFAS Function 2.c): An operator action to back-up this signal is not included in the PRA model; therefore, this F&O has no impact on the proposed changes.

Reactor Building Pressure High 3 (ESFAS Function 3.b.2): An operator action to back-up this signal is not included in the PRA model; therefore, this F&O has no impact on the proposed changes.

Emergency Feedwater Suction Transfer on Low Pressure (ESFAS Function 6.h): An operator action to back-up this signal is not included in the PRA model; therefore, this F&O has no impact on the proposed changes.

RWST Level Low-Low (ESFAS Function 8.a): This action is required to mitigate Loss of Coolant Accident (LOCA) events or events that result in LOCAs. This action is required to achieve successful event mitigation and is not a backup to another operator action. To arrive at the point in the event mitigation when this action is required, the operators and plant would have had a number of successful safety system actuations. Therefore, the human error probability for this operator action is not dependent on previous operator failures and this F&O has no impact on the proposed changes.

DE-03: The following observations were made regarding the internal flooding analysis.

1. The internal flooding analysis, as documented in the Individual Plant Examination (IPE) Internal Flooding Analysis Notebook, included a number of assumptions, which are documented in Section 1.3 of the Internal Flooding Analysis Notebook. The set of assumptions is reasonable with the possible following exceptions:

   (a) Walls and doors are assumed to remain intact throughout the flooding event, and doors are assumed to remain intact and in their normal position. This is optimistic, and ignores the potential that non-water-tight doors could be failed by a rising water level, or that normally-closed doors might be inadvertently left open, allowing flood propagation to adjacent rooms/areas.

(b) The potential for propagation through drains (grates, openings between floors, etc.) or vent lines is not addressed in the assumptions, nor is the ultimate disposition of the water, although the room-by-room evaluation indicates that propagation was considered in the analysis. However, where propagation is considered, it reflects the assumption noted in item 1 above, i.e., doors are assumed to limit propagation potential perfectly.

Review of the room-by-room screening documentation in the flooding notebook indicates that potential flood propagation was considered for each area, although details of the evaluation are sometimes sketchy. The extent of propagation considered is limited by the use of the above assumptions, e.g., for some rooms, propagation is assumed to only be possible through the gaps under the doors, whereas additional propagation might be possible if failure of the doors was considered.

2. The IPE analysis makes assumptions regarding the status, and even the presence, of flood barriers. Since these assumptions are an integral part of the analysis, they should be confirmed as still applicable, (e.g., curbs still present).

3. The internal flooding analysis uses the existing transient accident scenarios to model the plant response to an internal flooding initiator, appropriately failing equipment identified as potentially affected by the initiator. However, it does not appear that flood scenario-specific consideration has been given to human actions that are incorporated into the selected transient models. Although many such actions would likely not be affected, it is important to evaluate to determine that each action is still possible given the flood effects that cues for action are not adversely affected by the flood, and that response times inherent in the existing HEPs are not significantly changed by the flood scenario.

Assessment of Impact

The subject of this F&O concerns flooding events. Specifically of concern is 1) the assumption that walls and doors remain intact throughout the flooding event, 2) the status of flood barriers, and 3) flood scenario-specific consideration for human actions. Since a generic analysis was done in WCAP-14333-P-A, Rev. 1, to demonstrate the acceptability of the proposed changes, this F&O has no impact on the signals addressed generically in the WCAP.

With regard to the Reactor Building Pressure High 3 (ESFAS Functions 2.c and 3.b.2), Emergency Feedwater Suction Transfer on Low Pressure (ESFAS Function 6.h), and RWST Level Low-Low (ESFAS Function 8.a) channels, the identified issues will not impact the proposed changes, since the availability of these signals will not be impacted by a flooding event. The identified flooding events at VCSNS are related to the loss of a component cooling water train or the loss of a service water train, which have no impact on actuation signal availability.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

South Carolina Electric & Gas Company (SCE&G) has evaluated the proposed changes to the VCSNS TS described above against the significant Hazards Criteria of 10CFR50.92 and has determined that the changes do not involve any significant hazard. The following is provided in support of this conclusion.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The same reactor trip system (RTS) and engineered safety feature actuation system (ESFAS) instrumentation will continue to be used. The protection systems will continue to function in a manner consistent with the plant design basis. These changes to the Technical Specifications do not result in a condition where the design, material, and construction standards that were applicable prior to the changes are altered.

The proposed changes will not modify any system interface. The proposed changes will not affect the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident. There will be no changes to normal plant operating parameters or accident mitigation performance. The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR.

The determination that the results of the proposed changes are acceptable was established in the NRC SE issued for WCAP-14333, dated July 15, 1998. Implementation of the proposed changes will result in an insignificant risk impact. The proposed changes to Action 16 of TS 3/4.3.2 are also acceptable as demonstrated by meeting the acceptance criteria contained in Regulatory Guides 1.174 and 1.177.

The proposed changes to the AOTs and bypass test times, reduce the potential for inadvertent reactor trips and spurious ESF actuations, and therefore do not increase the probability of any accident previously evaluated. The proposed changes do not change the response of the plant to any accidents and have an insignificant impact on the reliability of the RTS and ESFAS signals. The RTS and ESFAS will remain highly reliable and the proposed changes will not result in a significant increase in the risk of plant operation. This is demonstrated by showing that the impact on plant safety as measured by the increase in CDF is less than 1.0E-06 per year and the increase in LERF is less than 1.0E-07 per year. In addition, for the AOT and bypass test time changes, the ICCDP and ICLERP values are less than 5.0E-07 and 5.0E-08,

respectively. The proposed changes meet the acceptance criteria in Regulatory Guides 1.174 and 1.177. Therefore, since the RTS and ESFAS will continue to perform their functions with high reliability as originally assumed, and the increase in risk as measured by the ΔCDF, ΔLERF, ICCDP, ICLERP risk metrics is within the acceptance criteria of Regulatory Guides 1.174 and 1.177, there will not be a significant increase in the consequences of any accidents.

The proposed changes to the bypass test times and AOTs do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event to within the applicable acceptance criteria. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed changes are consistent with safety analysis assumptions and resultant consequences.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

There are no hardware changes or any changes in the method by which any safety-related plant system performs its safety function. The proposed changes will not affect the normal method of plant operation. No performance requirements will be affected or eliminated. The proposed changes will not result in a physical alteration to any plant system or a change in the method by which any safety-related plant system performs its safety function. There will be no setpoint changes or changes to accident analysis assumptions.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of these changes.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Response: No.

The proposed changes do not affect the acceptance criteria for any analyzed event nor is there a change to any Safety Analysis Limit (SAL). There will be no effect on the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the DNBR limits, FQ, FΔH, LOCA PCT, peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria continue to be met.

Redundant RTS and ESFAS trains are maintained, and diversity with regard to the signals that provide reactor trip and engineered safety features actuation is also maintained. All signals credited as primary or secondary, and all operator actions credited in the accident analyses will remain the same. The proposed changes will not result in plant operation in a configuration outside the design basis. The calculated impact on risk is insignificant and meets the acceptance criteria contained in Regulatory Guides 1.174 and 1.177. Although there was no attempt to quantify any positive human factors benefit due to increased AOTs and bypass test times, it is expected that there would be a net benefit due to the reduced potential for spurious reactor trips and actuations associated with testing and maintenance activities.

Implementation of the proposed changes is expected to result in an overall improvement in safety, as follows:

- Improvements in the effectiveness of the operating staff in monitoring and controlling plant operation will be realized. This is due to less frequent distraction of the operators and shift supervisor to attend to RTS and ESFAS instrumentation Actions with short AOTs.

- The increased AOTs will provide more time for trouble shooting and repair activities, therefore reducing the potential for spurious trips and actuations.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Pursuant to 10CFR50.91, the preceding analyses provide a determination that the proposed Technical Specification changes pose no significant hazard as delineated by 10CFR50.92.

5.2 Applicable Regulatory Requirements/Criteria

The regulatory bases and guidance documents associated with the systems discussed in this amendment application include:

GDC 2 requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without the loss of the capability to perform their safety functions.

GDC 4 requires that structures, systems, and components important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with the normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, discharging fluids that may result from equipment failures, and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

GDC-13 requires that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems.

GDC-20 requires that the protection system(s) shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

GDC-21 requires that the protection system(s) shall be designed for high functional reliability and testability.

GDC-22, GDC-23, GDC-24, GDC-25, and GDC-29 require various design attributes for the protection system(s), including independence, safe failure modes, separation from control systems, requirements for reactivity control malfunctions, and protection against anticipated operational occurrences.

10CFR50.55a(h)(2) requires that the protection systems meet IEEE 279-1971.

Regulatory Guide 1.22 discusses an acceptable method of satisfying GDC-20 and GDC-21 regarding the periodic testing of protection system actuation functions. These periodic tests should duplicate, as closely as practicable, the performance that is required of the actuation devices in the event of an accident.

There will be no changes to the RTS or ESFAS instrumentation design, therefore, compliance with any of the regulatory requirements and guidance documents discussed above is not impacted. The evaluations previously discussed confirm that the plant will continue to comply with all applicable regulatory requirements.

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

5.2.1 Design Bases (FSAR)

FSAR Sections 7.2, "REACTOR TRIP SYSTEM," and 7.3, "ENGINEERED SAFETY FEATURES ACTUATION SYSTEM"

The VCSNS FSAR is unaffected by the proposed changes.

5.2.2 Approved Methodologies

The proposed changes do not result in a change to any methodologies.

5.2.3 Analysis

The analyses that support the changes contained in WCAP-14333-P-A, Rev. 1, are applicable to VCSNS. The changes to Action 16 of TS 3/4.3.2 are justified based on a risk-informed analysis performed in accordance with Regulatory Guides 1.174 and 1.177.

5.2.4 Conclusion

The proposed changes are consistent with WCAP-14333-P-A, Rev. 1, and TSTF-418, Rev. 2, which are both approved by the NRC. The proposed changes to Action 16 of TS 3/4.3.2 are acceptable as demonstrated by meeting the acceptance criteria contained in Regulatory Guides 1.174 and 1.177.

6.0 ENVIRONMENTAL CONSIDERATION

SCE&G has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10CFR20, or would change an inspection or surveillance requirement. SCE&G has evaluated the proposed changes and has determined that the changes do not involve, (i) a significant hazards consideration, (ii) a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. As discussed above, the proposed changes do not involve a significant hazards consideration. Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10CFR51, specifically 10CFR51.22(c)(9). Therefore, pursuant 10CFR51.22(b), an environmental assessment of the proposed changes is not required.

7.0 REFERENCES

1. VCSNS License Amendment No. 101, "Reactor Trip System Instrumentation and Engineered Safety Features Actuation System," dated June 18, 1991.

2. Correction to VCSNS License Amendment No. 101, dated August 6 1991.

3. VCSNS License Amendment No. 167, "Revision to Allowed Outage Time for Engineered Safety Features Actuation System Instrumentation Channels."

4. Southern Nuclear Operating Company letter LCV-1364, "Vogtle Electric Generating Plant Request to Revise Technical Specifications Reactor Trip System and Engineered Safety Features Actuation System Completion Times and Bypass Test Times," dated October 13, 1999 and LCV-1364-A, "Vogtle Electric Generating Plant Request to Revise Technical Specifications Reactor Trip System and Engineered Safety Features Actuation System Completion Times and Bypass Test Times Response to Request for Additional Information," dated June 1, 2000, Docket Numbers 50-424 and 50-425.

5. Westinghouse Owners Group letter, OG-96-110, "Transmittal of Response to Request for Additional Information (RAI) Regarding WCAP-14333-P Entitled 'Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times'," dated December 20, 1996.

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

Attachment to License Amendment No. XXX
To Facility Operating License No. NPF-12
Docket No. 50-395

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3/4 3-6	3/4 3-6
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
3/4 3-23	3/4 3-23
3/4 3-24	3/4 3-24
B 3/4 3-1	B 3/4 3-1
B 3/4 3-1a	B 3/4 3-1a
B 3/4 3-1b	B 3/4 3-1b
B 3/4 3-1c	B 3/4 3-1c
new	B 3/4 3-1d

SCE&G -- EXPLANATION OF CHANGES

Page	Affected Section	Bar #	Description of Change	Reason for Change
3/4 3-6	Action 2a.	1	Changed AOT from 6 hours to 72 hours.	Implementation of WCAP-14333-P-A, Rev. 1, and TSTF-418, Rev. 2.
	Action 2b.	2	Changed bypass test time from 4 hours to 12 hours.	Implementation of WCAP-14333-P-A, Rev. 1, and TSTF-418, Rev. 2.
3/4 3-7	Action 6a.	1	Changed AOT from 6 hours to 72 hours.	Implementation of WCAP-14333-P-A, Rev. 1, and TSTF-418, Rev. 2.
	Action 6b.	2	Change bypass test time from 4 hours to 12 hours.	Implementation of WCAP-14333-P-A, Rev. 1, and TSTF-418, Rev. 2.

Page	Affected Section	Bar #	Description of Change	Reason for Change
3/4 3-8	Action 8	1	Changed Action (Insert 1) to restore an inoperable RTB to operable status in 1 hour. Changed Action (Insert 2) that one RTB may be bypassed for up to 2 hours for maintenance on the undervoltage or shunt trip mechanisms. Changed Action so that an RTB may be bypassed for up to 4 hours for concurrent surveillance testing of the RTB and automatic trip logic	Insert 1 - Implement changes contained in NUREG-1431, Rev. 3. Insert 2 - Implementation of WCAP-14333-P-A, Rev. 1, and TSTF-418, Rev. 2.
	Action 10	2	Added comma after "Channels"	Editorial Correction.
	Action 11	3	Added "to" to end of sentence.	Editorial Correction.
	Action 12	4	Made two editorial changes - added a dash after ACTION 12 and capitalized "Minimum". Changed AOT from 6 hours to 24 hours.	Editorial Correction. Implementation of WCAP-14333-P-A, Rev. 1, and TSTF-418, Rev. 2.
3/4 3-23	Action 14	1	Changed AOT from 6 hours to 24 hours.	Implementation of WCAP-14333-P-A, Rev. 1, and TSTF-418, Rev. 2.
	Action 16	2	Changed AOT from 6 hours to 72 hours.	Plant specific Regulatory Guide 1.174 and 1.177 analysis.
	Action 16	3	Changed AOT bypass test time from 4 hours to 12 hours.	Plant specific Regulatory Guide 1.174 and 1.177 analysis.
3/4 3-24	Action 21	1	Changed "channels" to "channel". Changed AOT from 6 hours to 24 hours.	Editorial correction. Implementation of WCAP-14333-P-A, Rev. 1, and TSTF-418, Rev. 2.

Page	Affected Section	Bar #	Description of Change	Reason for Change
3/4 3-24	Action 24a.	2	Changed AOT from 6 hours to 72 hours.	Implementation of WCAP-14333-P-A, Rev. 1, and TSTF-418, Rev. 2.
	Action 24b.	3	Changed AOT bypass test time from 4 hours to 12 hours.	Implementation of WCAP-14333-P-A, Rev. 1, and TSTF-418, Rev. 2.
	Action 25	4	Changed "channels" to "channel". Changed AOT from 6 hours to 24 hours.	Editorial correction. Implementation of WCAP-14333-P-A, Rev. 1, and TSTF-418, Rev. 2.
B3/4 3-1	B3/4.3.1 B3/4.3.2	1 2 3	Modified verbage. Insert 1 - Added reference to WCAP-14333-P-A, Rev. 1. Insert 2 - Added reference to plant specific Regulatory Guide 1.174 and 1.177 analysis and Tier 2 requirement.	Implementation of WCAP-14333-P-A, Rev. 1, and plant specific Regulatory Guide 1.174 and 1.177 analysis. Note: Applicable Amendment numbers have been moved to the appropriate pages.
B3/4 3-1a B3/4 3-1b B3/4 3-1c	B3/4.3.1 B3/4.3.2	1	Continuation from Insert 2.	Continuation and Repagination Only. Note: Applicable Amendment numbers have been moved to the appropriate pages.
B3/4 3-1d	B3/4.3.1 B3/4.3.2		New Page.	Repagination Only. Note: Applicable Amendment numbers have been moved to the appropriate pages.

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

#The provisions of Specification 3.0.4 are not applicable.

##Below the P-6 (Intermediate Range Neutron Flux Interlock) setpoint.

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

****Values left blank pending NRC approval of 2 loop operation.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours. 72

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1. 12

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) setpoint but below 10 percent of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10 percent of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours; and 72

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1. 12

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within one hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

INSERT 1

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

INSERT 2

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 10 - With the number of OPERABLE Channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 11 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

ACTION 12 - With the number of OPERABLE Channels one less than the minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within ~~6~~ 24 hours or be in at least HOT STANDBY within the next ~~6~~ 24 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

Insert 1

restore the inoperable channel to OPERABLE status within 1 hour or

Insert 2

, one channel may be bypassed for up to 2 hours for maintenance on the undervoltage or shunt trip mechanisms, provided the other channel is OPERABLE, and one channel may be bypassed for up to 4 hours for concurrent surveillance testing of the Reactor Trip Breaker and automatic trip logic, provided the other channel is OPERABLE

TABLE 3.3-3 (Continued)

TABLE NOTATION

\# Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) setpoint.

\## Trip function may be blocked in this MODE below the P-12 (Low-Low T_{avg} Interlock) setpoint.

\### Except when below P-12 with all MSIVs and bypasses closed and disabled.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within ~~6~~ 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - DELETED

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may continue provided the inoperable channel is placed in bypass and the Minimum Channels OPERABLE requirement is met. Restore the inoperable channel to OPERABLE status within ~~6~~ 72 hours otherwise;

Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

One additional channel may be bypassed for up to ~~4~~ 12 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

ACTION 18 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 20 - With less than the Minimum Number of Channels OPERABLE, within one hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 21 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel~~s~~ to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

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ACTION 22 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 24 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours. 72

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1. 12

ACTION 25 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable Channel~~s~~ to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

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3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Protection System and Engineered Safety Feature Actuation System Instrumentation and interlocks ensure that 1) the associated action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoints, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and, 3) sufficient system functions capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses. The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance ~~and surveillance and maintenance outage times~~ intervals have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for Reactor Protection Instrumentation System," and supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

Insert 1

Insert 2

The Engineered Safety Feature Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A setpoint is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the setpoints have been specified in Table 3.3-4. Operation with setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error.

The methodology to derive the trip setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the trip setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this

Inserts for Bases Section B3/4.3.1 and B3/4.3.2

Insert 1

Specified surveillance and maintenance outage times have been determined in accordance with WCAP-14333-P-A, Rev. 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," and Westinghouse letter CGE-05-46.

Insert 2

Consistent with the requirement in Regulatory Guide 1.177 to include Tier 2 insights into the decision-making process before taking equipment out of service, restrictions on concurrent removal of certain equipment when a logic train is inoperable for maintenance are included (note that these restrictions do not apply when a logic train is being tested under the 4-hour bypass Note). Entry into Actions 12, 14, 21, or 25 is not a typical, pre-planned evolution during power operation, other than for surveillance testing. Since Actions 12, 14, 21, or 25 are typically entered due to equipment failure, it follows that some of the following restrictions may not be met at the time of entry into Actions 12, 14, 21, or 25. If this situation were to occur during the 24-hour AOT of Actions 12, 14, 21, or 25, the configuration risk assessment procedure will assess the emergent condition and direct activities to restore the inoperable logic train and exit Actions 12, 14, 21, or 25, or fully implement these restrictions, or perform a unit shutdown, as appropriate from a risk management perspective. The following restrictions will be observed:

- To preserve ATWS mitigation capability, activities that degrade the availability of the emergency feedwater system, RCS pressure relief system (pressurizer PORVs and safety valves), AMSAC, or turbine trip should not be scheduled when a logic train is inoperable for maintenance.

- To preserve LOCA mitigation capability, one complete ECCS train that can be actuated automatically must be maintained when a logic train is inoperable for maintenance.

- To preserve reactor trip and safeguards actuation capability, activities that cause master relays or slave relays in the available train to be unavailable and activities that cause analog channels to be unavailable should not be scheduled when a logic train is inoperable for maintenance.

- Activities on electrical systems (e.g., AC and DC power) and cooling systems (e.g., service water and component cooling water) that support the systems or functions listed in the first three bullets should not be scheduled when a logic train is inoperable for maintenance. That is, one complete train of a function that supports a complete train of a function noted above must be available.

Repagination Only

REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (continued)

will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the reactor trip and the engineered safety feature actuation associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Response time may be verified by actual response time tests in any series of sequential, overlapping, or total channel measurements, or by the summation of allocated sensor, signal processing, and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise or power interrupt tests), (2) in place, onsite, or offsite (e.g., vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

WCAP-14036-P-A, Revision 1, "Elimination of Periodic Response Time Tests," provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning, and actuation logic response times must be verified prior to placing the component into operational service and re-verified following maintenance or modification that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for the repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing element of a transmitter.

Westinghouse letter CGE-00-018, dated March 28, 2000, provided an evaluation of the Group 05 (11NLP and 6NSA) 7300 process cards. These cards were revised after the submittal of WCAP-14036, Revision 1. This letter concluded that the failure modes and effects analysis (FMEA) performed for the older versions of these cards and documented in WCAP-14036-P-A, Revision 1, is applicable for these Group 05 cards. The bounding time response values determined by test and evaluation and reported in the WCAP are valid for these redesigned cards.

REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (continued)

The Engineered Safety Features response times specified in Table 3.3-5 which include sequential operation of the RWST and VCT valves (Notes 2 and 3) are based on values assumed in the non-LOCA safety analyses. These analyses are for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction isolation valves are closed following opening of the RWST charging pumps suction valves. When the sequential operation of the RWST and VCT valves is not included in the response times (Note 1) the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response times specified in Table 3.3-5 will assure that the assumptions used for the LOCA and non-LOCA analyses with respect to the operation of the VCT and RWST valves are valid.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those engineered safety features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss of coolant accident 1) safety injection pumps start and automatic valves position, 2) reactor trip, 3) feedwater isolation, 4) startup of the emergency diesel generators, 5) containment spray pumps start and automatic valves position, 6) containment isolation, 7) steam line isolation, 8) turbine trip, 9) auxiliary feedwater pumps start and automatic valves position, 10) containment cooling fans start and automatic valves position, 11) essential service water pumps start and automatic valves position, and 12) control room isolation and ventilation systems start.

Several automatic logic functions included in this specification are not necessary for Engineered Safety Feature System actuation but their functional capability at the specified setpoints enhances the overall reliability of the Engineered Safety Features functions. These automatic actuation systems are purge and exhaust isolation from high containment radioactivity, turbine trip and feedwater isolation from steam generator high-high water level, initiation of emergency feedwater on a trip of the main feedwater pumps, automatic transfer of the suctions of the emergency feedwater pumps to service water on low suction pressure, and automatic opening of the containment recirculation sump suction valves for the RHR and spray pumps on low-low refueling water storage tank level.

Repagination Only

REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (continued)

The service water response time includes: 1) the start of the service water pumps and, 2) the service water pumps discharge valves (3116A,B,C-SW) stroking to the fully opened position. This condition of the valves assures that flow will become established through the component cooling water heat exchanger, diesel generator coolers, HVAC chiller, and to the suction of the service water booster pumps when these components are placed in-service. Prior to this time, the flow is rapidly approaching required flow and sufficient pressure is developed as valves finish their stroke. Each of the above-listed components will be starting to perform their accident mitigation function, either directly or indirectly depending upon the use of the component, and will be operational within the service water response time of 71.5/81.5 seconds[1/]. Only the service water booster pumps have a direct impact on the accident analysis via the RBCUs' heat removal capability as discussed below.

The RBCU response time includes: 1) the start of the RBCU fan and the service water booster pumps and, 2) all the service water valves which must be driven to the fully opened or fully closed position. This condition of the valves allows the flow to become fully established through the RBCU. Prior to this time, the flow is rapidly approaching required flow as the valves finish their stroke. Although the RBCU would be removing heat throughout the Engineered Safety Features response time, the accident analysis does not assume heat removal capability from 0 to 71.5 seconds [2/] because the industrial cooling water system is not completely isolated until 71.5 seconds. A linear ramp increase from 95% full heat removal capability to 100% full heat removal capability is assumed by the accident analysis to start at 71.5 seconds and end at 86.5 seconds[3/]. Full heat removal capability is assumed at 86.5 seconds based on the position of the valve 3107-SW.

1/ Total time is 1.5 second instrument response after setpoint is reached, plus 10 seconds diesel generator start, plus 10 seconds to reach service water pump start and begin 3116-SW opening via Engineered Safety Features Loading Sequencer, plus 60 seconds stroke time for 3116-SW. During this total time, the service water pumps start and the service water pump discharge valve begins to open at 11.5 seconds and the pump discharge valve is fully open at 71.5 seconds without a diesel generator start required and 21.5 seconds and 81.5 seconds including a diesel generator start.

2/ Total time is 1.5 second instrument response after setpoint Is reached, plus 10 second diesel start plus 60 seconds* for valves to isolate industrial cooling water system.

3/ Total time is 1.5 second instrument response after setpoint is reached, plus 10 second diesel generator start plus 75 seconds to stroke valves 3107A, B-SW.

* During this time period, the Engineered Safety Features Loading Sequencer starts the RBCU fans at 25 seconds and service water booster pumps at 30 seconds after the valves begin to stroke.

ATTACHMENT II

PROPOSED TECHNICAL SPECIFICATION CHANGES (RETYPED)

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

\# The provisions of Specification 3.0.4 are not applicable.

\## Below the P-6 (Intermediate Range Neutron Flux Interlock) setpoint.

\### Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

**** Values left blank pending NRC approval of 2 loop operation.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 72 hours.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.3.1.1.

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) setpoint but below 10 percent of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10 percent of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 72 hours; and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within one hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE, one channel may be bypassed for up to 2 hours for maintenance on the undervoltage or shunt trip mechanisms, provided the other channel is OPERABLE, and one channel may be bypassed for up to 4 hours for concurrent surveillance testing of the Reactor Trip Breaker and automatic trip logic, provided the other channel is OPERABLE.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 10 - With the number of OPERABLE Channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 11 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

ACTION 12 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

TABLE 3.3-3 (Continued)

TABLE NOTATION

Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) setpoint.

Trip function may be blocked in this MODE below the P-12 (Low-Low T_{avg} Interlock) setpoint.

Except when below P-12 with all MSIVs and bypasses closed and disabled.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - DELETED

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may continue provided the inoperable channel is placed in bypass and the Minimum Channels OPERABLE requirement is met. Restore the inoperable channel to OPERABLE status within 72 hours otherwise;

Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

One additional channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

ACTION 18 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 20 - With less than the Minimum Number of Channels OPERABLE, within one hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 21 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 22 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 24 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 72 hours.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 25 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable Channel to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Protection System and Engineered Safety Feature Actuation System Instrumentation and interlocks ensure that 1) the associated action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoints, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and, 3) sufficient system functions capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses. The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance intervals have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for Reactor Protection Instrumentation System," and supplements to that report. Specified surveillance and maintenance outage times have been determined in accordance with WCAP-14333-P-A, Rev. 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," and Westinghouse letter CGE-05-46. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

Consistent with the requirement in Regulatory Guide 1.177 to include Tier 2 insights into the decision-making process before taking equipment out of service, restrictions on concurrent removal of certain equipment when a logic train is inoperable for maintenance are included (note that these restrictions do not apply when a logic train is being tested under the 4-hour bypass Note). Entry into Actions 12, 14, 21, or 25 is not a typical, pre-planned evolution during power operation, other than for surveillance testing. Since Actions 12, 14, 21, or 25 are typically entered due to equipment failure, it follows that some of the following restrictions may not be met at the time of entry into Actions 12, 14, 21, or 25. If this situation were to occur during the 24-hour AOT of Actions 12, 14, 21, or 25, the configuration risk assessment procedure will assess the emergent condition and direct activities to restore the inoperable logic train and exit Actions 12, 14, 21, or 25, or fully implement these restrictions, or perform a unit shutdown, as appropriate from a risk management perspective. The following restrictions will be observed:

- To preserve ATWS mitigation capability, activities that degrade the availability of the emergency feedwater system, RCS pressure relief system (pressurizer PORVs and safety valves), AMSAC, or turbine trip should not be scheduled when a logic train is inoperable for maintenance.

- To preserve LOCA mitigation capability, one complete ECCS train that can be actuated automatically must be maintained when a logic train is inoperable for maintenance.

REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (continued)

- To preserve reactor trip and safeguards actuation capability, activities that cause master relays or slave relays in the available train to be unavailable and activities that cause analog channels to be unavailable should not be scheduled when a logic train is inoperable for maintenance.

- Activities on electrical systems (e.g., AC and DC power) and cooling systems (e.g., service water and component cooling water) that support the systems or functions listed in the first three bullets should not be scheduled when a logic train is inoperable for maintenance. That is, one complete train of a function that supports a complete train of a function noted above must be available.

The Engineered Safety Feature Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A setpoint is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the setpoints have been specified in Table 3.3-4. Operation with setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error.

The methodology to derive the trip setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the trip setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the reactor trip and the engineered safety feature actuation associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Response time may be verified by actual response time tests in any series of sequential, overlapping, or total channel measurements, or by the summation of allocated sensor, signal processing, and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained

REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (continued)

from: (1) historical records based on acceptable response time tests (hydraulic, noise or power interrupt tests), (2) in place, onsite, or offsite (e.g., vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

WCAP-14036-P-A, Revision 1, "Elimination of Periodic Response Time Tests," provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning, and actuation logic response times must be verified prior to placing the component into operational service and re-verified following maintenance or modification that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for the repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing element of a transmitter.

Westinghouse letter CGE-00-018, dated March 28, 2000, provided an evaluation of the Group 05 (11NLP and 6NSA) 7300 process cards. These cards were revised after the submittal of WCAP-14036, Revision 1. This letter concluded that the failure modes and effects analysis (FMEA) performed for the older versions of these cards and documented in WCAP-14036-P-A, Revision 1, is applicable for these Group 05 cards. The bounding time response values determined by test and evaluation and reported in the WCAP are valid for these redesigned cards.

The Engineered Safety Features response times specified in Table 3.3-5 which include sequential operation of the RWST and VCT valves (Notes 2 and 3) are based on values assumed in the non-LOCA safety analyses. These analyses are for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction isolation valves are closed following opening of the RWST charging pumps suction valves. When the sequential operation of the RWST and VCT valves is not included in the response times (Note 1) the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response times specified in Table 3.3-5 will assure that the assumptions used for the LOCA and non-LOCA analyses with respect to the operation of the VCT and RWST valves are valid.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those engineered safety features components whose

REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (continued)

aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss of coolant accident 1) safety injection pumps start and automatic valves position, 2) reactor trip, 3) feedwater isolation, 4) startup of the emergency diesel generators, 5) containment spray pumps start and automatic valves position, 6) containment isolation, 7) steam line isolation, 8) turbine trip, 9) auxiliary feedwater pumps start and automatic valves position, 10) containment cooling fans start and automatic valves position, 11) essential service water pumps start and automatic valves position, and 12) control room isolation and ventilation systems start.

Several automatic logic functions included in this specification are not necessary for Engineered Safety Feature System actuation but their functional capability at the specified setpoints enhances the overall reliability of the Engineered Safety Features functions. These automatic actuation systems are purge and exhaust isolation from high containment radioactivity, turbine trip and feedwater isolation from steam generator high-high water level, initiation of emergency feedwater on a trip of the main feedwater pumps, automatic transfer of the suctions of the emergency feedwater pumps to service water on low suction pressure, and automatic opening of the containment recirculation sump suction valves for the RHR and spray pumps on low-low refueling water storage tank level.

The service water response time includes: 1) the start of the service water pumps and, 2) the service water pumps discharge valves (3116A,B,C-SW) stroking to the fully opened position. This condition of the valves assures that flow will become established through the component cooling water heat exchanger, diesel generator coolers, HVAC chiller, and to the suction of the service water booster pumps when these components are placed in-service. Prior to this time, the flow is rapidly approaching required flow and sufficient pressure is developed as valves finish their stroke. Each of the above-listed components will be starting to perform their accident mitigation function, either directly or indirectly depending upon the use of the component, and will be operational within the service water response time of 71.5/81.5 seconds[1/]. Only the service water booster pumps have a direct impact on the accident analysis via the RBCUs' heat removal capability as discussed below.

[1/] Total time is 1.5 second instrument response after setpoint is reached, plus 10 seconds diesel generator start, plus 10 seconds to reach service water pump start and begin 3116-SW opening via Engineered Safety Features Loading Sequencer, plus 60 seconds stroke time for 3116-SW. During this total time, the service water pumps start and the service water pump discharge valve begins to open at 11.5 seconds and the pump discharge valve is fully open at 71.5 seconds without a diesel generator start required and 21.5 seconds and 81.5 seconds including a diesel generator start.

REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (continued)

The RBCU response time includes: 1) the start of the RBCU fan and the service water booster pumps and, 2) all the service water valves which must be driven to the fully opened or fully closed position. This condition of the valves allows the flow to become fully established through the RBCU. Prior to this time, the flow is rapidly approaching required flow as the valves finish their stroke. Although the RBCU would be removing heat throughout the Engineered Safety Features response time, the accident analysis does not assume heat removal capability from 0 to 71.5 seconds [2/] because the industrial cooling water system is not completely isolated until 71.5 seconds. A linear ramp increase from 95% full heat removal capability to 100% full heat removal capability is assumed by the accident analysis to start at 71.5 seconds and end at 86.5 seconds[3/]. Full heat removal capability is assumed at 86.5 seconds based on the position of the valve 3107-SW.

[2/] Total time is 1.5 second instrument response after setpoint Is reached, plus 10 second diesel start plus 60 seconds* for valves to isolate industrial cooling water system.

[3/] Total time is 1.5 second instrument response after setpoint is reached, plus 10 second diesel generator start plus 75 seconds to stroke valves 3107A, B-SW.

* During this time period, the Engineered Safety Features Loading Sequencer starts the RBCU fans at 25 seconds and service water booster pumps at 30 seconds after the valves begin to stroke.

ATTACHMENT III
LIST OF REGULATORY COMMITMENTS

The following actions are committed to by SCE&G in this document.

- Activities that degrade the availability of the Emergency Feedwater system, RCS pressure relief system (Pressurizer PORVs and safety valves), AMSAC, or Turbine trip should not be scheduled when a logic train is inoperable for maintenance.
- One complete ECCS train that can be actuated automatically must be maintained when a logic train is inoperable for maintenance.
- Activities that cause master relays or slave relays in the available train to be unavailable and activities that cause analog channels to be unavailable should not be scheduled when a logic train is inoperable for maintenance.
- Activities on electrical systems (e.g., AC and DC power) and cooling systems (e.g., Service Water and Component Cooling Water) that support the systems or functions listed in the first three bullets above should not be scheduled when a logic train is inoperable for maintenance. That is, one complete train of a function that supports a complete train of a function noted above must be available.