December 20, 2002

Mr. Robert H. Bryan, Chairman
Westinghouse Owners Group
Tennessee Valley Authority
Mail Code LP4J-C
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT WCAP-15376-P, REV. 0, "RISK-INFORMED ASSESSMENT OF THE RTS AND ESFAS SURVEILLANCE TEST INTERVALS AND REACTOR TRIP BREAKER TEST AND COMPLETION TIMES" (TAC. NO. MB0983)

Dear Mr. Bryan:

By letter dated November 8, 2000, as supplemented by letters dated June 8, June 25, and September 28, 2001, and January 8, 2002, the Westinghouse Owners Group (WOG) submitted the subject topical report (TR) prepared by Westinghouse Electric Company, LLC, that revises the technical specifications for the reactor trip system and engineered safety features actuation system instrumentation. The proposed changes include increasing the completion time and bypass time for the reactor trip breakers, as well as the surveillance test intervals for the reactor trip breakers, master relays, logic cabinets, and analog channels. The proposed changes adopt the staff's approved Technical Specification Task Force (TSTF) Traveler TSTF-411, Rev. 1, "Surveillance Test Interval Extension for Components of the Reactor Protection System," submitted by letter dated August 9, 2001.

The NRC staff has completed its review of the subject TR. The TR is acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the report and in the associated NRC safety evaluation (SE), which is enclosed. The enclosed SE defines the basis for acceptance of the TR.

The staff has concluded that the proposed generic TS changes are consistent with the approved allowances for testing with an instrument channel in bypass and for repair completion times accepted by the staff based on WCAP-15376-P. In addition, proposed TS Bases provide an adequate basis or reason for the standard technical specification (STS) changes. Therefore, Westinghouse should include TSTF-411, Rev. 1, with publication of the approved version of WCAP-15376-P. Licensees may then propose to adopt the approved TS during a conversion to the STS or as a separate license amendment application for WCAP-15376-P.

Pursuant to 10 CFR 2.790, we have determined that the enclosed SE does not contain proprietary information. However, we will delay placing the SE in the public document room for ten working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any information in the enclosure is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.790.

We do not intend to repeat our review of the matters described in the subject report, and found acceptable, when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. Our acceptance applies only to matters approved in the report.

In accordance with the procedures established in NUREG-0390, the NRC requests that the WOG publish an accepted version within three months of receipt of this letter. The accepted version shall incorporate (1) this letter and the enclosed SE between the title page and the abstract, (2) all requests for additional information from the staff and all associated responses, and (3) a "-A" (designating "accepted") following the report identification symbol.

Should our criteria or regulations change so that our conclusions as to the acceptability of the report are invalidated, the WOG and/or the licensees referencing the TR will be expected to revise and resubmit their respective documentation, or submit justification for the continued applicability of the TR without revision of their respective documentation.

Sincerely,

/RA/

William H. Ruland, Director
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Safety Evaluation

cc w/encl:
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/RA/
William H. Ruland, Director
Project Directorate IV
Division of Licensing Project Management
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Enclosure: Safety Evaluation

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

WCAP-15376-P, REV 0, "RISK-INFORMED ASSESSMENT OF THE RTS AND ESFAS

SURVEILLANCE TEST INTERVALS AND REACTOR TRIP BREAKER

TEST AND COMPLETION TIMES"

WESTINGHOUSE OWNERS GROUP

PROJECT NO. 694

1.0 INTRODUCTION

By letter dated November 8, 2000, and its supplemental letters dated June 8, June 25, September 28, 2001, and January 8, 2002, the Westinghouse Owners Group (WOG) submitted WCAP-15376-P, Rev. 0, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times." WCAP-15376-P, Rev. 0, provides justification for increasing the allowed outage time (AOT)/completion time (CT) and bypass times for the reactor trip breaker (RTB), as well as the surveillance test interval (STI) for the RTB, master relays, and logic cabinets.

The proposed changes adopt the Nuclear Energy Institute (NEI) Technical Specifications Task Force (TSTF) Traveler TSTF-411, Rev. 1, "Surveillance Test Interval Extension for Components of the Reactor Protection System," submitted by letter dated August 9, 2001.

The CT is defined as part of the limiting condition for operation (LCO) in the improved standard technical specifications (STSs). The AOT is a general reference to time to accomplish a technical specification (TS) required Action. To have more specific meaning, AOT can refer to additional time for repair, bypass, shutdown, etc. A CT has a broader meaning than an AOT, by also defining the time for other required actions such as equipment status or plant mode changes. The CT is intended to allow sufficient time to repair failed equipment while minimizing the risk associated with the loss of the component function.

The purpose of the program is to provide the technical justification for extending the STI for components for the reactor protection system. The components specifically included are analog channels, logic cabinets, master relays, and reactor trip breakers. This program also provides the technical justification for extending the RTB completion time (allowed outage time) for one RTB inoperable to 24 hours from 1 hour and the bypass time for an RTB to 4 hours from 2 hours. This safety evaluation considers both the solid state protection system (SSPS) and relay protection system. An extension of the STI reduces the required testing on the reactor protection system components without significantly impacting its reliability, and reduces the potential for reactor trips and actuations of engineered safety features associated with the testing of these components. An extension of the CT increases the unavailability of a component due to the increased time the component is down for maintenance. The CT risk is reflected in the core damage frequency (CDF) and the large early release frequency (LERF) by adjusting the component unavailability due to maintenance. The CT extensions for the RTB will provide the licensees additional time to complete test and maintenance activities while at power, potentially reducing the number of forced outages related to compliance with reactor trip breaker CTs, and provide consistency with the CTs for the logic cabinets. For CTs, the designated CTs may not provide adequate time for repair, but longer CTs may incur a relatively larger risk. Note that the STS replaced the term AOT with CT, which has a broader meaning than AOT by also defining the time for other required actions such as equipment status or plant mode changes.

By contrast, STIs are intervals for surveillance tests scheduled periodically as required by the TS. Such tests are performed to ensure that safety-related equipment continues to be operable and failures are detectable, thereby limiting the fault exposure time. The primary risk contribution attributed to increasing an STI comes from the increased probability of a component failure between scheduled STIs and, therefore, the probability that the component will be inoperable during the surveillance interval. The extension of an STI affects the yearly risk, which is represented by the CDF and LERF. An STI extension can affect the yearly risk in several ways:

- Reduce the risk by decreasing the number of test-caused reactor trips by limiting the opportunity for test-caused errors. This occurs simply because increasing the STI decreases the amount of testing for a given time.

- Reduce the risk by decreasing the unavailability of the reactor protection system (RPS) component by reducing the test frequency.

- Increase the risk by increasing the fault exposure time as described above. This is attributable to the fact that the increased STI increases the interval during which the equipment is subject to failure during standby. As the fault exposure time increases, there is a greater probability that failures during standby will not be detected for RPS components involved with the STI extension.

For an STI, the idea is to strike a balance between more frequent testing (which can adversely impact safety either through errors during testing, spurious actuations, misconfiguration, or equipment wearout) and extended intervals (which can increase fault exposure times). The designated CTs may not provide adequate time for repair, but longer CTs may incur a relatively larger risk. A risk-informed approach to CTs and STIs in conjunction with engineering evaluations, can provide insights that allow CTs and STIs to be optimized without significantly increasing plant risk.

The NRC's policy statement on the use of probabilistic risk assessment (PRA) methods in nuclear regulatory activities encourages the use of PRA to improve safety-related decision-making and regulatory efficiency. Under this policy, the NRC staff may use traditional engineering analysis, as well as risk-informed approaches, to evaluate licensee-initiated licensing changes that go beyond current staff positions. In WCAP-15376-P, Rev. 0, the WOG stated that the proposed changes to the STIs will reduce the required testing on RPS components without significantly impacting the reliability of the reactor trip system (RTS), while reducing the potential for reactor trips and actuation of engineered safety features associated with the testing of these components. The WOG also stated that extending the CTs for the RTBs will provide additional time to complete test and maintenance activities while at power, and provide consistency with the CT for the logic cabinets.

The proposed increases in STIs, CTs, and bypass times for both the SSPS and relay protection system RTS and associated engineered safety features actuation system (ESFAS) designs are as follows:

(1) SOLID STATE PROTECTION SYSTEM

- Surveillance Test Intervals
  - Logic cabinet: From 2 months to 6 months
  - Master Relay: From 2 months to 6 months
  - Analog Channels: From 3 months to 6 months
  - Reactor Trip Breaker: From 2 months to 4 months
- Completion Time
  - Reactor Trip Breakers: From 1 hour to 24 hours
- Bypass Times
  - Reactor Trip Breakers: From 2 hours to 4 hours

(2) RELAY PROTECTION SYSTEM

- Surveillance Test Intervals
  - Logic Cabinet: From 1 month to 6 months
  - Master Relay: No change
  - Analog Channels: From 3 months to 6 months
  - Reactor Trip Breakers: From 2 months to 4 months
- Completion Time
  - Reactor Trip Breakers: From 1 hour to 24 hours
- Bypass Time
  - Reactor Trip Breakers: From 2 hours to 4 hours

Whereas the CT is the additional time that is available to correct a fault that is discovered during testing and the bypass time is defined as the amount of time a component can be bypassed for surveillance testing.

Depending on the plant protection system design, some of the actuation logic and master relays associated with the containment purge and exhaust isolation instrumentation (STS 3.3.6) and control room emergency filtration system (CREFS) actuation instrumentation (STS 3.3.7) TSs may be processed through the relay or solid state protection system. Since the STIs for the actuation logic and master relays of the ESFAS Instrumentation were justified to be relaxed in this report, these STI relaxations are also applicable to the actuation logic and master relays for all signals processed through the relay or SSPS.

The STI for the source range neutron flux channel operational test (COT) in the RTS instrumentation (STS 3.3.1) TS was justified to be relaxed in this report. Since this source range neutron flux channel is also used for the boron dilution protection system (BDPS) (STS 3.3.9), the STI relaxation is also applicable to that STI.

The approach used in this program is consistent with the NRC's approach for using PRA in risk-informed decisions on plant-specific changes to the current licensing basis as presented in Regulatory Guides (RGs) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," and 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specification." The approach addresses the impact on defense-in-depth and the impact on safety margins, as well as an evaluation of the impact on risk. The risk evaluation considers the three-tiered approach as presented in RG 1.177 for the extension to the RTB CT. Tier 1, PRA Capability and Insights, assesses the impact of the proposed CT (AOT) change on CDF, incremental conditional core damage probability (ICCDP), LERF, and incremental conditional large early release probability (ICLERP). Tier 2, Avoidance of Risk-Significant Plant Configurations, considers potential risk-significant plant operating configurations. Tier 3, Risk-Informed Plant Configuration Control and Management, will be addressed on a plant-specific basis when the TS CT change is implemented by each licensee.

2.0 REGULATORY EVALUATION

The NRC staff formed a task group in August 1983 to investigate problems and recommend improvements concerning surveillance testing required by TS. The results of the Task Group study were published in November 1983 in NUREG-1024, "Technical Specifications-Enhancing the Safety Impact." NUREG-1024 recommended that the staff (1) review the bases for TS test frequencies, (2) ensure that the TS required tests promote safety and do not degrade equipment, and (3) review surveillance tests to ensure that they do not unnecessarily burden personnel.

The technical specification improvement program (TSIP) was established in December 1984 to provide the framework for addressing the recommendations of NUREG-1024, and for rewriting and improving the STS. The results of the TSIP were documented in NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements." The TSIP study concluded that, while some testing at power is essential, safety can be improved, equipment degradation decreased, and unnecessary personnel burden prevented by reducing the amount of testing performed at power.

In 1983, the WOG submitted WCAP-10271-P, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," which provided a methodology to be used to justify revisions to a plant's TS. The WOG Technical Specification Optimization Program (TOP) evaluated changes to surveillance test intervals and allowed outage times for the analog channels, logic cabinets, master and slave relays, and reactor trip breakers. The methodology evaluated increasing surveillance intervals, increases in test and maintenance out-of-service times and bypassing portions of the RPS during test and maintenance. The WOG stated in WCAP-10271-P that plant staff devote significant time and effort to perform, review, document, and track surveillance activities that, in many instances, may not be required on the basis of the high reliability of the equipment. The justification for the changes was the small impact that the changes would have on plant risk.

In WCAP-10271-P, the WOG performed fault tree analyses to calculate the reactor trip unavailability considering surveillance intervals and test and maintenance times. The sensitivity to variations in surveillance intervals and test and maintenance times was also evaluated with respect to maintaining or revising current surveillance intervals. The WOG concluded that the results of the analyses for the RPS were adequate to justify a revision of the STS. The staff accepted WCAP-10271-P by safety evaluation report (SER), with provisions, dated February 21, 1985, in which the staff approved the following changes for plant-specific TS:

1. Increase the surveillance interval for RTS analog channel operational tests from once per month to once per quarter.

2. Increase the time in which an inoperable RTS analog channel may be maintained in an untripped condition from 1 hour to 6 hours.

3. Increase the time an inoperable RTS analog channel may be bypassed to allow testing of another channel in the same function from 2 hours to 4 hours. Also, the channel test may be done in the bypass mode leaving the inoperable channel in tripped condition.

4. Allow testing of the RTS analog channels in a bypass condition instead of a tripped condition.

Subsequent to the approval of WCAP-10271-P, the WOG submitted WCAP-14333-P, “Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times,” dated May 1995. The purpose of this WCAP was to evaluate the following changes to the TS:

1. Increase the bypass times and the CTs for both the solid state and relay protection system RPS and ESFAS designs: (i) for the analog channels the CT increased from 6 hours to 72 hours, and the bypass time from 4 hours to 12 hours, and (ii) for the logic cabinets, master and slave relay CTs were increased from 6 hours to 24 hours.

2. Revise the action statement for an inoperable slave relay to increase the CT for maintenance to 24 hours, with an additional 6 hours for the mode change.

3. For cases where the logic cabinets and the trip breakers both cause their train to be inoperable when in test or maintenance, allow the reactor trip breakers to be bypassed for the period of time equivalent to the bypass time for the logic cabinets, provided that both are tested at the same time.

The staff approved WCAP-14333-P by SER dated July 15, 1998, subject to the condition that licensees confirm the applicability of the WCAP to their plant, and that licensees address RG 1.177, Tier 2 and Tier 3 analysis, including the incorporation of applicable Configuration Risk Management Program (CRMP) insights.

To facilitate the implementation of risk-informed methodology, general guidance for evaluating the technical basis for proposed changes is provided in Chapter 19.0 of the Standard Review Plan (SRP). More specific guidance related to risk-informed TS changes is provided in Section 16.1 of the SRP. Chapter 19.0 of the SRP states that a risk-informed application should be evaluated to ensure that the proposed changes meet the following key principles:

1. The proposed change meets the current regulations, unless it explicitly relates to a requested exemption or rule change.

2. The proposed change is consistent with the defense-in-depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes increase core damage frequency or risk, the increase should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

5. The impact of the proposed change should be monitored using performance measurement strategies.

With respect to the above principles for risk-informed licensing basis changes, RG 1.174 and RG 1.177 identify a four-element approach for use in evaluating a plant's design, operations, and other activities associated with evaluating risk-informed regulatory changes:

Element 1: Define the Proposed Change

When defining the proposed change, a requested TS change may be acceptable if it (1) improves operational safety, (2) can be supported on the basis of risk implications, and/or (3) reduces unnecessary regulatory burden.

Element 2: Perform an Engineering Analysis

RG 1.174 states that the technical basis for the proposed change should be rooted in traditional engineering and system analysis. The proposed TS change should not be based solely on PRA results.

Element 3: Define Implementation and Monitoring Program

The licensee should develop and define a CRMP. This program is to be used to ensure that risk-significant plant configurations will not be entered, and appropriate actions are available if unforeseen events put the plant in a risk-significant configuration. The CRMP should ensure that an extension of a TS CT or STI does not degrade operational safety over time. Additionally, the licensee's Maintenance Rule program should ensure that when equipment does not meet its performance criteria, an evaluation of the equipment associated with the CT or STI will be performed.

Element 4: Submit Proposed Change

The proposed TS change should be documented and included in the licensee's amendment request, and should include risk-informed TS change documentation showing that the objectives of the NRC's PRA policy statement are being met and are consistent with the key principles and elements of RGs 1.174 and 1.177.

As part of Element 2, RG 1.177 identifies a three-tiered approach for a licensee to evaluate the risk associated with a proposed TS change:

- Tier 1 is an evaluation of the plant-specific risk associated with the proposed TS change, as shown by the change in CDF and ICCDP. Where applicable, containment performance should be evaluated on the basis of an analysis of LERF and ICLERP.

- Tier 2 identifies and evaluates, with respect to defense-in-depth, any potential risk-significant plant equipment outage configurations associated with the proposed change. The licensee should provide reasonable assurance the risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed TS change is out-of-service.

- Tier 3 provides for the establishment of an overall CRMP and confirmation that its insights are incorporated into the decisionmaking process before taking equipment out-of-service prior to or during the CT. Compared to Tier 2, Tier 3 provides additional coverage on the basis of any additional risk-significant configurations that may be encountered during maintenance scheduling over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule, 10 CFR 50.65(a)(4), which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance, testing, and corrective and preventive maintenance.

On February 6, 1987, the Commission issued guidelines for improving the content and quality of nuclear power plant TS, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (52 FR 3788). During the period from 1989 to 1992, utility owners groups and the staff developed improved STS that would establish models of the Commission's policy for each primary reactor type.

In September 1992, the Commission issued Revision 0 of the improved STS as NUREGs 1430-1434, which were developed using the guidance and criteria contained in the Commission's Interim Policy Statement. The ISTS reflect the results of a detailed review of the application of the interim policy statement criteria to generic system functions, which were published in a "Split Report" issued to the nuclear steam supply system (NSSS) vendor owners groups in May 1988.

In June 2001, Revision 2 of NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," was published. The changes to Revision 1 that are reflected in Revision 2 resulted from the experience gained from license amendment applications to convert to these improved STS or to adopt partial improvements to existing technical specifications. NUREG-1431, Revision 2 is the result of extensive public technical meetings and discussions between the

NRC staff and various nuclear power plant licensees, NSSS vendors owners groups and the NEI TSTF.

The review of proposed generic changes to Westinghouse STS (NUREG-1431) is a multi-staged process designed to ensure that each STS remains internally consistent, maintains coherence among the various vendor's STS, and incorporates the knowledge and operating experience of the industry and the NRC. Changes to the STS NUREGs, which are potentially applicable to multiple plants, are proposed to the NRC by the NEI sponsored TSTF through publicly available submittals. The TSTF includes representatives from the four U.S. commercial nuclear power plant owner groups and NEI. The NRC staff reviews the changes to the STS proposed by the TSTF (referred to as TSTF changes) and will accept, modify, or reject them. Once TSTF changes are accepted, they are considered to be part of the STS. Individual licensees may propose to adopt the TSTF changes during a conversion to the STS or as a separate license amendment application.

The TSTF process facilitates licensees adopting NRC-accepted changes to the STS for their specific plant TS. This process is intended to streamline the license amendment review process involving NRC-accepted STS changes in order to increase NRC efficiency and reduce unnecessary regulatory burden. The NRC role in maintaining plant safety is achieved by the technical review of proposed changes to the STS as well as plant-specific applications to adopt NRC-accepted changes to the STS.

3.0 TECHNICAL EVALUATION

The WOG stated that the approach used in WCAP-15376-P, Rev. 0, to justify the proposed revisions to CTs and STIs for the RTS and ESFAS, is consistent with the guidance outlined in RGs 1.174 and 1.177. The WOG further stated that the increase in surveillance intervals will reduce the required testing on the reactor protection system components without significantly reducing their reliability, and reduce the potential for reactor trips and actuation of engineered safety features associated with testing of these components. In addition, the WOG stated that the CT extensions for the reactor trip breakers will provide the licensees additional time to complete test and maintenance activities while at power, potentially reducing the number of forced outages related to compliance with reactor trip breakers and provide consistency with the CT previously approved by the staff for the logic cabinets under WCAP-14333-P. The staff used a three-tiered approach in its evaluation of the risk associated with the proposed TS changes in RPS and ESFAS surveillance test, completion, and bypass times. The review approach is consistent with the guidance in RG 1.177. The first tier evaluates the PRA model and includes the RTS and ESFAS unavailability analyses and risk analyses that support the risk impact assessment. The second tier addresses the need to preclude potentially high risk configurations should additional equipment outages occur during the proposed CT period. The third tier evaluates the licensee's configuration risk management program to ensure that equipment outage due to maintenance, testing, or random failure immediately prior to or during the proposed CT will be appropriately assessed from a risk perspective.

3.1 Tier 1: PRA Capability and Insights

Westinghouse used traditional PRA methodology to evaluate the requested TS changes. To support this assessment, two aspects had to be considered: (1) an evaluation of the PRA model and application to the proposed changes, and (2) an evaluation of PRA results and insights stemming from the application. The staff concluded that Westinghouse's PRA is valid for assessing the proposed TS changes and identifies the impact of the TS change on plant risk. The WOG stated that the unavailability data used in the model came from several sources including previous RTS and ESFAS studies, WCAP-10271 and WCAP-14333-P. The WOG also used data from NUREG/CR-5500, Vol. 2, "Reliability Study: Westinghouse Reactor Protection System, 1984-1995."

The staff's review concerned itself with the development of the PRA model and its applicability in the evaluation of plant risk based on the proposed changes. Westinghouse used component failure probabilities derived from NUREG/CR-5500, Vol. 2, and additional component failure probabilities from WCAP-10271-P, and WCAP-14333-P, both of which were previously approved by the staff. The WOG also surveyed various plants to obtain operational data for SSPS safeguard driver cards and master relays for both the relay and SSPS-based RPS. As a result, the failure probabilities used in WCAP-15376-P, Rev. 0, were developed using plant operating experience rather than the generic reliability factors used in WCAP-10271-P and WCAP-14333-P.

The plant survey data indicated that the failure probability of the master relay for the relay protection system was higher than the SSPS. Based on this, the WOG chose not to propose extending STIs for the master relays associated with a relay protection system, but maintain surveillance testing at current intervals.

3.1.1 Evaluation of PRA Model and Its Application to the CT Extension

WCAP-15376-P, Rev. 0, used the Vogtle PRA model to evaluate the impact on risk of the proposed changes. The Vogtle PRA was developed in response to Generic Letter (GL) 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities." The staff concluded that the Vogtle Individual Plant Examination (IPE) met the intent of GL 88-20. The Vogtle PRA model was previously utilized in WCAP-14333-P to provide the basis tor extending CTs for the RPS. Since the requested surveillance interval and CT requests in WCAP-15376-P, Rev. 0, are similar in scope to those requested by WCAP-14333-P, the WOG utilized the same model for WCAP-15376-P, Rev. 0. WCAP-15376-P, Rev. 0, provides various insights as to the appropriateness of the Vogtle model in support of the proposed TS changes stating that the model provides sufficient detail to perform the analysis, the Vogtle model includes anticipated transient without scram (ATWS), and the Vogtle model allows operator action to be credited.

The model used in WCAP-14333-P is not identical to the model used for WCAP-15376-P, Rev. 0. Changes were made to the model including the replacement of WCAP-10271 data with proprietary plant data collected by Westinghouse and the use of NUREG/CR-5500 failure data. Logic changes included the modeling of the SSPS at the card level instead of the component level as was done in WCAP-10271-P and WCAP-14333-P. The staff did not review the quality of the proprietary data in detail, but the use of more recent generic data, including Westinghouse specific data, should result in improved assessment of the unavailability estimates. Based on the use of the same model as a previous evaluation (with updated data), the staff finds that the quality of the PRA is sufficient for the evaluation of the proposed changes. However, the analysis did not report uncertainty bounds for the proprietary data estimates which may have an influence on plant-specific results. Based on the above, a plant-specific analysis should consider the uncertainty in the data consistent with RG 1.174 and RG 1.177 guidance to ensure that the conclusions of WCAP-15376-P, Rev. 0, remain valid for the plant-specific case.

The analysis performed in WCAP-15376-P, Rev. 0, included fault trees of representative RTS signals and ESFAS actuation for the SSPS (including 2 of 3 and 3 of 4 logic) and relay protection system (including 2 of 3 and 3 of 4 logic). WCAP-15376-P, Rev. 0, concluded that the SSPS unavailability estimates bound the relay protection system unavailability estimates. As a result, the estimates for the SSPS were used in the analysis of the Vogtle PRA in estimating the CDF and LERF for the current TS case and the proposed TS surveillance intervals and CTs.

To evaluate the results presented in WCAP-15376-P, Rev. 0, an independent model was developed for selected RPS signals. In response to the staff's request, the WOG provided additional data to quantify the base case. The generic data that was used in the model was verified. Revised component unavailabilities and failure probabilities were used to determine new signal unavailabilities. The proposed TS amendment request changes were individually evaluated using the same cases as WCAP-15376-P, Rev. 0. The "combined cases" representing the proposed TS changes were evaluated for the bounding SSPS plant. The results were consistent with those reported in WCAP-15376-P, Rev. 0.

3.1.2 Change in CDF

The unavailability analysis did not model or evaluate all RTS and ESFAS signals in the fault tree analysis. Consistent with WCAP-14333-P, only representative signals were evaluated in detail. The risk analysis used the results from the unavailability analysis to determine the impact that the proposed changes had on the availability of the RPS. The base case was represented by the CTs, bypass times, and STIs previously approved in WCAP-14333-P. The representative signals included safety injection, auxiliary feedwater, reactor trip initiation - pressurizer high, and reactor trip on pressurizer high or over temperature delta T. The availability of diverse signals, including operator action if the automatic actuation fails, were considered in the WCAP-15376-P, Rev. 0 analysis. The representative model selected was based on the Vogtle plant which is a variation on the model used for the analysis in WCAP-14333-P and included fault trees for both RPS and ESFAS. The fault trees were modeled in sufficient detail to allow the CTs and STI to be varied for the components included in WCAP-15376, Rev. 0. The base case model was quantified according to the approved changes in WCAP-14333-P and includes updated component data using Westinghouse proprietary and generic plant failure rates. The WCAP-15376-P, Rev. 0 analysis did not credit any potential trip reduction over that taken by the previous WCAP-10271-P study. WCAP-15376-P, Rev. 0 took credit for decreased unavailability due to reduced test frequency and accounted for the increase in fault exposure time when increasing the STI intervals.

The baseline value for CDF was calculated to be 5.05E-5/r-yr for both 2/4 logic and 2/3 logic RPS. The topical report then presented a series of TS sensitivity cases with each case including RPS components slated for STI or CT modification and the CDF and LERF calculated and compared to the acceptance guidelines defined by RG 1.174 and RG 1.177. For the proposed TS changes, the CDF increased to 5.13E-5/r-yr and 5.14E-5/r-yr for 2/4 and 2/3 logic signals, respectively. RG 1.174 acceptance guidelines state that when the calculated increase in CDF is less than 10E-6/r-yr, the change will be considered regardless of whether there is calculation of total CDF as long as there is no indication that the total CDF is not considerably higher than 10E-4/r-yr. For increases in CDF in the range of 10E-6/r-yr to 1E-5/r-yr, applications will be considered only if it can be reasonably shown that the total CDF is less than 10E-4/r-yr. Therefore, the proposed CT and STI changes are acceptable to the staff with respect to ΔCDF.

The WOG also evaluated the LERF for the RTBs, master relays, logic cabinets, and analog channels. Based on the values presented in WCAP-15376-P, Rev. 0, the change in LERF when implementing proposed TS changes is 3.09E-8/r-yr and 5.68E-8/r-yr for 2/4 and 2/3 logic signals, respectively. Both values are within the 10E-7/r-yr of acceptance guidelines stated in RG 1.174 and are acceptable to the staff.

These values are based on the assumption that the only contributions to LERF would come from containment bypass events and core damage events with the containment not isolated. The contributions from containment failure events are not considered in WCAP-15376-P, Rev. 0 based on the Vogtle PRA and the assumption that Vogtle is representative of all Westinghouse plants. There may be exceptions to this assumption, including Westinghouse plants with an ice condenser containment. Studies have shown that ice condenser plants can be substantially more sensitive to early containment failure than pressurized water reactors (PWRs) with a large dry or sub-atmospheric containment. For example, in an ice condenser plant with a higher station blackout frequency, early containment failure may be important. A plant-specific assessment of containment failures should be performed for all plants referencing this topical report to determine whether there are any impacts on the proposed TS changes. WCAP-15376-P, Rev. 0 also states that if the ICCDP value meets the RG criterion then the ICLERP value would also meet the associated guideline value in the RG. These assumptions may not be the case for specific plants in that the plant differences may affect the results when compared to the reference plant. Therefore, for plant-specific cases, a licensee will need to confirm that both the ICCDP and ICLERP values for the proposed change meet the guidance outlined in RG. 1.177 and RG 1.174.

3.1.3 Single CT Risk

The base case model was also re-quantified to evaluate the proposed CT for the RTB. The model assumed one RTB was out-of-service with the associated bypass breaker available. The operable RTB and the in-service bypass breaker provide the reactor trip. In this arrangement both breakers are controlled by the logic cabinet associated with the operable breaker. The proposed change revises the RTB bypass time to 4 hours to be consistent with the logic cabinets and the CT for the RTBs also is increased to 24 hours to match the logic cabinet CT. The WOG estimated a conditional CDF of 7.07E-5/r-yr for this configuration and estimated the ICCDP to be 6.9E-8/r-yr. The value for the ICCDP is within the RG 1.177 acceptance guideline of less than 5.0E-7/r-yr.

However, WCAP-14333-P accepted the case where a logic cabinet and associated RTB may be tested concurrently, provided that the RTB is bypassed for a period of time equivalent to the bypass time for the logic cabinet. This testing arrangement causes the respective RPS train to be inoperable when in a test or maintenance condition. Because WCAP-14333-P approved concurrent testing of the RTB and associate logic cabinet, the staff questioned modeling the proposed bypass time and CT with only the RTB out of service. The WOG's response indicated that the WOG's intent is to remove both the RTB and its associated logic cabinet from service during surveillance testing. With the more limiting configuration of having both the RTB and the associated logic cabinet out-of-service, the conditional CDF was calculated to be 1.45E-4/r-yr with an ICCDP risk of 3.2E-7/r-yr. The risk of this configuration is substantially higher (by a factor of 5) than when only an RTB is inoperable, but is more representative of the LCO configuration to be implemented during surveillance testing. However, the revised ICCDP still remains bounded by the RG 1.177 acceptance guideline of 5.0E-7/r-yr. The change in CDF also meets RG 1.174 acceptance guidelines (see Section 3.1.2). A licensee implementing this surveillance configuration may require additional plant-specific Tier 1 and Tier 2 analyses to confirm that the generic analysis for WCAP-15376-P, Rev. 0, remains bounding for the plant-specific case.

3.1.4 Shutdown Risk and Transition Risk

WCAP-15376-P, Rev. 0, states that one advantage for extending the CT for the RTBs is that the exposure to transition risk would be decreased since the extended CT would limit the transition to lower modes should the present RTB CT be exceeded. The WOG also claimed that the transition risk would be comparable to the risk increase caused by the requested CT extension for the RTBs.

The staff finds that the evaluation of transition risk would only occur when unscheduled corrective maintenance cannot be completed within the allotted time specified by the TS. In cases where a failure condition is observed during an RTB surveillance test, the decision to repair at power or perform a mode change should consider the transition risk. However, it has limited applicability to the proposed surveillance AOT extension request. The analysis presented in WCAP-15376-P, Rev. 0 for maintenance risk and transition risk assumes only that the RTB is out-of-service and not a complete train of RPS.

3.1.5 Common Cause Failures

The WOG used the Multiple Greek Letter Method (MGL) for the analog channels and the Bete Factor approach for the RTB, logic cabinet, master and slave relays. The analysis in WCAP-15376-P, Rev. 0 did not distinguish between components being down due to failure (corrective maintenance) when evaluating common cause failures. In response to the request for additional information (RAI), the WOG provided an estimate of the single CT risk for both corrective maintenance and preventive maintenance. Based on the WOG's results, the single CT risk did not change for corrective or preventive maintenance. The WOG stated that a significant change in risk is not observed since the reactor trip signal is dominated by the failure of the logic cabinet as opposed to the failure of both RTBs. These results are only applicable for surveillance performed with both the RTB and logic cabinet out-of-service. In this case, the remaining operable logic cabinet failures appear to dominate the failure of the RPS signal since the logic cabinet supports both RPS and ESFAS functions. For cases where only an RTB is removed, then the unaffected RTB may become risk significant.

3.1.6 Application of Vogtle Model to the Plant-Specific Case

The applicability of the Vogtle PRA model to other Westinghouse plants was evaluated by the staff. WCAP-15376-P, Rev. 0 states that RPS/ESFAS functions are similar in response across all Westinghouse plants for initiating events. Additionally, the safety functions challenged in response to initiating events and the associated actuation signals generated are also similar and procedures provide for operator action to back up automatic initiation of safety systems.

Although the staff recognizes the similarity between plant RPS and ESFAS systems, design, function, and initiating event frequency, the unavailability of the RPS shows a wide range of estimates. These differences may result from varying model assumptions (including operator action), the generic or plant-specific data used, actual design differences or variations in plant-specific equipment performance (master relays for example). Another example identified in the review was what appeared to be a substantial variability in the contribution to core damage due to ATWS events. The WOG provided a summary of ATWS contributions for various plants. Based on the data provided, the contribution to core damage frequency for ATWS events at Westinghouse plants varied from less than 0.1 to approximately 20 percent with the WCAP-15376-P, Rev. 0 Vogtle model showing a contribution of 2.1 percent. Another factor that may contribute to the variability in plant risk is the assumption of operator action in the PRA model. The analysis in WCAP-15376-P, Rev. 0 is centered on the automatic functions performed by the RPS with operator action credited in the topical report. Based on the above, a licensee incorporating WCAP-15376-P, Rev. 0 is expected to confirm the applicability of the topical report to their plant and to address any design or performance differences that may affect the proposed CT and STI assumptions. Additionally, to ensure consistency with the reference plant, the model assumptions for human reliability in the topical report should be confirmed to be applicable to the plant-specific case. In the Tier 1 evaluation for WCAP-15376-P, Rev. 0, the WOG evaluated the impact of the proposed changes on CDF, ICCDP, LERF, and ICLERP. The staff found that the use of the Vogtle PRA as a representative model was reasonable for assessing the proposed TS changes and that the risk impact was within the guidelines stated for ΔCDF, ICCDP, ΔLERF, and ICLERP in RG 1.174 and RG 1.177. However, the applicability of the generic model must be confirmed when applying the results of WCAP-15376-P, Rev. 0 to a plant-specific license amendment.

The WOG stated that although the WCAP-15376-P, Rev. 0 analysis and the results obtained were only for analog systems, the results are also applicable to digital systems based on previous applications of WOG TOP with Eagle 21 systems. The staff notes that the Eagle 21 system provides for improved on-line monitoring and based on previous evaluations has similar unavailabilities to an analog RTS. However, the Eagle 21 upgrade only replaced the channel process logic modules of the RTS with an integrated microprocessor-based module and thus was limited in scope. Digital upgrades with increased scope, integration, and architectural differences may affect plant risk and therefore surveillance requirements. Therefore, the staff finds that the generic applicability of WCAP-15376-P, Rev. 0 to future digital systems is not clear and should be considered on a plant-specific basis.

3.2 Impact on Defense-In-Depth and Safety Margins

The traditional engineering considerations need to be addressed. These include defense-in-depth and safety margins. The fundamental safety principles on which the plant design is based cannot be compromised. Design basis accidents are used to develop the plant design. These are a combination of postulated challenges and failure events that are used in the plant design to demonstrate safe plant response. Defense-in-depth, the single failure criterion, and adequate safety margins may be impacted by the proposed change and consideration needs to be given to these elements.

3.2.1 Impact on Defense-In-Depth

The proposed STI changes to the RTS and ESFAS and the proposed change to the RTB CT have only a small calculated impact on CDF and LERF. The CT and STI changes to the RTB only impact CDF and have no impact on containment integrity. The STI changes to the analog channels, logic cabinets, and master relays have small calculated impacts on both CDF and LERF. These changes do not degrade core damage prevention at the expense of containment integrity, nor do these changes degrade containment integrity at the expense of 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 an accident or transient is not impacted. No new activities on the RPS will be performed at power that could lead to potentially new transient events. Conversely, the increase in STIs could potentially lead to a reduction in the likelihood of a test induced transient or accident.

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

There is no impact on the redundancy, independence, or diversity of the RPS or the ability of the plant to respond to events with diverse systems. The RPS is a diverse and redundant system and will remain so. There will be no change to the signals available to trip the reactor or initiate ESF functions. The RPS is a reliable system and is backed up by the plant operators who will still be available to perform actions in the occurrence of RPS failure. In addition, the RTS is backed up by ATWS mitigating system actuation circuitry (AMSAC) signal to start auxiliary feedwater and trip the turbine in conjunction with RCS pressure mitigation via the pressurizer safety valves and relief valves. The proposed changes have no impact on this alternate approach to ATWS mitigation.

Defense against common cause failures was reviewed by the staff. The extensions requested are not sufficiently long to expect new common cause failure mechanisms to arise. In addition, the operating environment for these components remains the same, so new common cause failure modes are not anticipated. Also, backup systems and operator actions are not impacted by these changes; and there are no common cause links between the RPS and these backup options. Furthermore, the RTB CT and bypass time increases are not requested to perform additional tests and routine maintenance activities while at power. Such activities will continue to be completed as currently required. Therefore, no new potential common cause failure mechanisms have been introduced.

No new operator actions related to the STI extension or the CT extension are required. No additional operating, maintenance, or test procedures have been introduced or modified due to these changes, and no new at-power tests or maintenance activities are expected to occur as a result of these changes. The plant will continue to be operated and maintained as before. With the CT increase, the plant can be maintained at power longer to complete repair activities on the RTBs. With the STI increase, fewer surveillance tests will need to be completed at-power which will reduce the potential for test induced reactor trips and safety system actuations.

3.2.2 Impact on Safety Margins

The safety analysis acceptance criteria as stated in the Final Safety Analysis Report is not impacted by these changes. Redundant RPS trains will be maintained. Diversity with regard to signals to provide reactor trip and actuation of engineered safety features will also be maintained. The proposed changes will not allow plant operation in a configuration outside the design basis. All signals credited as primary or secondary and all operator actions credited in the accident analysis will remain the same.

3.3 Tier 2: Avoidance of Risk-Significant Plant Configuration

The licensee should provide reasonable assurance that risk significant plant equipment outage configurations will not occur when specific plant equipment is out-of-service in accordance with the proposed TS change. The WOG identified the following restrictions on equipment removal when an RTB is out of service:

1. With an RTB out-of-service, systems designed to mitigate an ATWS event should be available. Also identified were RCS pressure relief, auxiliary feedwater flow, AMSAC, and turbine trip. Based on the above, WCAP-15376-P, Rev. 0 stated that activities that degrade the availability of auxiliary feedwater, RCS pressure relief, AMSAC, or turbine trip should not be scheduled when an RTB is out-of-service.

2. Because there is increased dependence on the available reactor trip train when one logic cabinet is removed from service, activities that could degrade other components of the RPS including master relays, slave relays, and analog channels should not be scheduled concurrently with a logic cabinet out of service.

3. WCAP-15376-P, Rev. 0 also noted that activities on electrical support systems for the equipment identified should not be scheduled during RTB maintenance.

Therefore, a licensee should evaluate the need for and develop the necessary restriction on concurrent equipment outages when entering proposed RTB CT to avoid potential risk significant configurations.

3.4 Tier 3: Risk-Informed Plant Configuration Control and Management

The WOG did not provide detailed information on the Tier 3, 10-CFR 50.65(a)(4) CRMP due to the plant-specific nature of the information required. Each licensee should develop a program that ensures that the risk impact of out-of-service equipment is appropriately evaluated prior to performing the maintenance activity. The program should be able to uncover risk significant plant outage configuration and should include such factors as equipment unavailability, operational activities, and weather conditions. The Tier 3 program provides additional assurance over the Tier 2 program by identifying risk significant configurations that may be encountered over extended periods of plant operation. The CRMP program referenced by RG 1.174 may be implemented by a licensee through the maintenance rule (10 CFR 50.65(a)(4)), which requires that the licensee before performing maintenance activities, shall assess and manage the increase in risk that may result from the proposed maintenance activity.

3.5 TSTF-411, Rev. 1 Evaluation

The proposed NUREG-1431 changes revise TSs and Bases for Reactor Protection System Instrumentation (3.3.1), Engineered Safety Feature System Instrumentation (3.3.2), Containment Purge and Exhaust Isolation Instrumentation (3.3.6), Control Room Emergency Filtration System Actuation Instrumentation (3.3.7), and Boron Dilution Protection System Instrumentation (3.3.9).

Specifically, the RTB bypass test time allowance changes to 4 hours from 2 hours; the CT allowance changes to 24 hours from 1 hour; and the surveillance frequency changes to 4 months from 2 months in Specification 3.3.1 for both SSPS and RPS designs. The surveillance frequencies for logic cabinets changes to 6 months from 2 months for SSPS plants and to 6 months from 1 month for RPS plants. Master relays changes to 6 months from 2 months for SSPS plants, and analog channels changes to 6 months from 3 months in Specifications 3.3.1, 3.3.2, 3.3.6, 3.3.7 and 3.3.9. In addition, changes were made to TS 3.3.1 and 3.3.2 Bases. Appropriate to WCAP-13632 and WCAP-14036, references were added to the Bases discussions in accordance with approved TSTF-111, Rev. 6. Also, references in Surveillance Requirement (SR) 3.3.7.3 and SR 3.3.9.3 were corrected to reflect an appropriate citation.

The NRC staff reviewed the proposed generic relaxations contained in TSTF-411, Rev. 1 and found them acceptable because they are consistent with current licensing practices and the Commission’s regulations.

3.5.1 <u>Relaxation of Completion Time</u>

Upon discovery of a failure to meet an LCO, TS specify times for completing Required Actions of the associated TS conditions. Required Actions establish remedial measures that must be taken within specified completion times. These times define limits during which operation in a degraded condition is permitted. Incorporating required action and completion time extensions is acceptable because these times take into account the operability status of the redundant systems of TS required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a design basis accident (DBA) occurring during the repair period.

The TSTF-411, Rev. 1 proposed changes reduce required testing on the reactor protection system components and reduce the potential for reactor trips and actuation of engineered safety features associated with the testing of these components. The required action CT extension for the RTBs will provide additional time to complete test and maintenance activities while at power, potentially reducing the number of forced outages related to compliance with RTB CTs, and provide consistency with the CTs for the testing of RPS logic cabinets.

3.5.2 Relaxation of Surveillance Requirement

TS require maintaining the LCO equipment operable by meeting the SRs in accordance with the specified SR Frequency. This requires conducting tests to demonstrate equipment is operable, or that LCO parameters are within specified limits. When the test acceptance criteria and any specified conditions for the conduct of the test are met, the equipment is deemed operable. TSTF-411, Rev. 1 includes changes related to relaxation of STS SR frequencies. Relaxing the SR frequency provides operational flexibility consistent with the objective of the STS without reducing confidence that the equipment is operable. The changes are acceptable because appropriate testing standards are retained for determining that the LCO-required features are operable. These relaxations of SRs optimize test requirements for the affected safety systems and increase operational flexibility.

4.0 CONCLUSION

The staff review of the proposed changes finds that WCAP-15376-P, Rev. 0 is consistent with acceptance guidelines of RG 1.174, RG 1.177, and staff guidance as outlined in NUREG-0800, "Standard Review Plan." From traditional engineering insights, including the defense-in-depth philosophy and the safety margins, the staff finds that the proposed changes have no impact on the defense-in-depth philosophy and safety margin. The staff further determines that the implementation of the proposed changes for CT and STI for RTS and ESFAS, including signals processed through either the relay or SSPS, should result in only a minimal quantitative impact on plant risk.

The staff also concludes that TSTF-411, Rev. 1 proposed generic TS changes are consistent with the approved allowances for RTB testing with an instrument channel in bypass, for RTB repair completion times and for surveillance frequency changes to logic cabinets for SSPS and relay protection system plant designs, for master relays for SSPS plants and analog channels accepted by the staff based on WCAP-15376-P, Rev. 0. In addition, the proposed TS Bases provide an adequate basis or reason for the STS changes and editorial guidelines of the STS "Writer's Guide" were followed for preparing STS changes. Thus, TSTF-411, Rev. 1 preserves the human factors principles used throughout the development of NUREG-1431 and can be appropriately applied to licensee specific TS changes.

5.0 CONDITIONS AND LIMITATIONS

Although the engineering consideration and PRA insights support the proposed changes, the applicability of WCAP-15376-P, Rev. 0 on a plant-specific basis needs to be confirmed by providing the following information:

1. A licensee is expected to confirm the applicability of the topical report to their plant, and to perform a plant-specific assessment of containment failures and address any design or performance differences that may affect the proposed changes.

2. Address the Tier 2 and Tier 3 analyses including risk significant configuration insights and confirm that these insights are incorporated into the plant-specific configuration risk management program.

3. The risk impact of concurrent testing of one logic cabinet and associated reactor trip breaker needs to be evaluated on a plant-specific basis to ensure conformance with the WCAP-15376-P, Rev. 0 evaluation, and RGs 1.174 and 1.177 guidance.

4. To ensure consistency with the reference plant, the model assumptions for human reliability in WCAP-15376-P, Rev. 0 should be confirmed to be applicable to the plant-specific configuration.

5. For future digital upgrades with increased scope, integration and architectural differences beyond that of Eagle 21, the staff finds the generic applicability of WCAP-15376-P, Rev. 0 to future digital systems not clear and should be considered on a plant-specific basis.

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