TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 6 months.*
E	At least once per 18 months.*
R	At least once per 24 months.*
S/U	Prior to each reactor startup.
N/A	Not applicable.

*In these Technical Specifications, 6 months is defined to be 184 days, 18 months is defined to be 550 days, and 24 months is defined to be 730 days.

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3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the Reactor Protection System instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each Reactor Protection System instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The total bypass function shall be demonstrated OPERABLE at least once per REFUELING INTERVAL during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME* of each reactor trip function shall be demonstrated to be within its limit at least once per REFUELING INTERVAL. Neutron detectors are exempt from response time testing; the response time of the neutron flux signal portion of the channel shall be measured from the neutron detector output or from the input of the first electronic component in the channel. Each test shall include at least one channel per function such that all channels are tested at least once every N times the REFUELING INTERVAL where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

* The response times include the sensor (except for the neutron detectors), Reactor Protection System instrument delay, and the control rod drive breaker delay. A delay time has been assumed for the Reactor Coolant Pump monitor in the determination of the response time of the High Flux/Number of Reactor Coolant Pumps On functional unit.

TABLE 4.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. High Flux	S	D(2), and Q(6,9)	N.A.	1, 2
3. RC High Temperature	S	R	SA(9)	1, 2
4. Flux - ΔFlux - Flow	S(4)	M(3) and Q(6,7,9)	N.A.	1, 2
5. RC Low Pressure	S	R	SA(9)	1, 2
6. RC High Pressure	S	R	SA(9)	1, 2
7. RC Pressure-Temperature	S	R	SA(9)	1, 2
8. High Flux/Number of Reactor Coolant Pumps On	S	Q(6,9)	N.A.	1, 2
9. Containment High Pressure	S	E	SA(9)	1, 2
10. Intermediate Range, Neutron Flux and Rate	S	E(6)	N.A.(5)	1, 2 and*
11. Source Range, Neutron Flux and Rate	S	E(6)	N.A.(5)	2, 3, 4 and 5
12. Control Rod Drive Trip Breakers	N.A.	N.A.	Q~~M~~(8,9) and S/U(1)(8)	1, 2 and*
13. Reactor Trip Module Logic	N.A.	N.A.	Q ~~M~~(9)	1, 2 and*
14. Shutdown Bypass High Pressure	S	R	SA(9)	2**,3**,4**,5**
15. SCR Relays	N.A.	N.A.	R	1, 2 and*

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TABLE 4.3-1 (Continued)

NOTATION

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - When THERMAL POWER [TP] is above 50% of RATED THERMAL POWER [RTP] and at a steady state, compare out-of-core measured AXIAL POWER IMBALANCE [API_o] to incore measured AXIAL POWER IMBALANCE [API_i] as follows:

$$\frac{RTP}{TP} [API_o - API_i] = \text{Offset Error}$$

Recalibrate if the absolute value of the Offset Error is $\geq$ 2.5%.

(4) - AXIAL POWER IMBALANCE and loop flow indications only.

(5) - CHANNEL FUNCTIONAL TEST is not applicable. Verify at least one decade overlap prior to each reactor startup if not verified in previous 7 days.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Flow rate measurement sensors may be excluded from CHANNEL CALIBRATION. However, each flow measurement sensor shall be calibrated at least once each REFUELING INTERVAL.

(8) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of both the undervoltage and shunt trip devices of the Reactor Trip Breakers.

(9) - Performed on a STAGGERED TEST BASIS.

* - With any control rod drive trip breaker closed.

** - When Shutdown Bypass is actuated.

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INSTRUMENTATION

ANTICIPATORY REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.3 The Anticipatory Reactor Trip System instrumentation channels of Table 3.3-17 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-17

ACTION: As shown in Table 3.3-17

SURVEILLANCE REQUIREMENTS

4.3.2.3 The Anticipatory Reactor Trip System shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST for the modes and at the frequencies shown in Table 4.3-17.

TABLE 4.3-17

ANTICIPATORY REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
1. Turbine Trip (a)	S	Not Applicable	SA (c)	1 (b)
2. Main Feed Pump Turbine Trip	S	Not Applicable	SA (c)	1
3. Output Logic	Not Applicable	Not Applicable	Q ~~M~~ (c)	1

(a) Trip automatically bypassed below 45 percent of RATED THERMAL POWER
(b) Applicable only above 45 percent of RATED THERMAL POWER
(c) Perform on a STAGGERED TEST BASIS

3/4.3 INSTRUMENTATION

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BASES

3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM AND SAFETY SYSTEM INSTRUMENTATION

The OPERABILITY of the RPS, SFAS and SFRCS instrumentation systems ensure that 1) the associated action and/or trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for RPS, SFAS and SFRCS purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance 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. The response time limits for these instrumentation systems are located in the Updated Safety Analysis Report and are used to demonstrate OPERABILITY in accordance with each system's response time surveillance requirements.

For the RPS, SFAS Table 3.3-4 Functional Unit Instrument Strings d and e and Interlock Channel a, and SFRCS Table 3.3-12 Functional Unit 2:

Only the Allowable Value is specified for each Function. Nominal trip setpoints are specified in the setpoint analysis. The nominal trip setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing are consistent with the assumptions of the specific setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis to account for instrument uncertainties appropriate to the trip parameter. These uncertainties are defined in the specific setpoint analysis.

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Setpoints must be found within the specified Allowable Values. Any setpoint adjustment shall be consistent with the assumptions of the current specific setpoint analysis.

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis.

The frequency is justified by the assumption of an 18 or 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(Next page is B3/4 3-1a)

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3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM AND SAFETY SYSTEM INSTRUMENTATION (Continued)

The measurement of response time at the specified frequencies provides assurance that the RPS, SFAS, and SFRCS action function associated with each channel is completed within the time limit assumed in the safety analyses.

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. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

The actuation logic for Functional Units 4.a., 4.b., and 4.c. of Table 3.3-3, Safety Features Actuation System Instrumentation, is designed to provide protection and actuation of a single train of safety features equipment, essential bus or emergency diesel generator. Collectively, Functional Units 4.a., 4.b., and 4.c. function to detect a degraded voltage condition on either of the two 4160 volt essential buses, shed connected loads, disconnect the affected bus(es) from the offsite power source and start the associated emergency diesel generator. In addition, if an SFAS actuation signal is present under these conditions, the sequencer channels for the two SFAS channels which actuate the train of safety features equipment powered by the affected bus will automatically sequence these loads onto the bus to prevent overloading of the emergency diesel generator. Functional Unit 4.a. has a total of four units, one associated with each SFAS channel (i.e., two for each essential bus). Functional Units 4.b. and 4.c. each have a total of four units, (two associated with each essential bus); each unit consisting of two undervoltage relays and an auxiliary relay.

An SFRCS channel consists of 1) the sensing device(s), 2) associated logic and output relays (including Isolation of Main Feedwater Non Essential Valves and Turbine Trip), and 3) power sources.

The SFRCS response time for the turbine stop valve closure is based on the combined response times of main steam line low pressure sensors, logic cabinet delay for main steam line low pressure signals and closure time of the turbine stop valves. This SFRCS response time ensures that the auxiliary feedwater to the unaffected steam generator will not be isolated due to a SFRCS low pressure trip during a main steam line break accident.

Safety-grade anticipatory reactor trip is initiated by a turbine trip (above 45 percent of RATED THERMAL POWER) or trip of both main feedwater pump turbines. This anticipatory trip will operate in advance of the reactor coolant system high pressure reactor trip to reduce the peak reactor coolant system pressure and thus reduce challenges to the pilot operated relief valve. This anticipatory reactor trip system was installed to satisfy Item II.K.2.10 of NUREG-0737. The justification for the ARTS turbine trip arming level of 45% is given in BAW-1893, October, 1985.

Docket Number 50-346
License Number NPF-3
Serial Number 2556
Enclosure 2

NRC LETTER
JANUARY 7, 1998
NRC EVALUATION OF B&WOG TOPICAL REPORT BAW-10167, SUPPLEMENT 3
JUSTIFICATION FOR INCREASING THE REACTOR TRIP SYSTEM
ON-LINE TEST INTERVALS

(11 pages follow)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 7, 1998

Mr. J. J. Kelly, Manager
Owners Group Services
Framatome Technologies, Inc.
P. O. Box 10935
Lynchburg, VA 24506-0935

SUBJECT: NRC EVALUATION OF B&WOG TOPICAL REPORT BAW-10167, SUPPLEMENT 3, JUSTIFICATION FOR INCREASING THE REACTOR TRIP SYSTEM ON-LINE TEST INTERVALS

Dear Mr. Kelly:

The purpose of this letter is to provide the enclosed safety evaluation report (SER) on the B&WOG Topical Report BAW-10167, Supplement 3 prepared by B&W Nuclear Technologies for the B&WOG Regulatory Reduction working group. This Topical Report was submitted to the NRC by letter dated June 7, 1996, and presents justification for extending the on-line surveillance test interval (STI) for reactor trip devices comprised of reactor trip breakers, reactor trip modules, and electronic trip relays from a one-month to a six-month interval. The B&WOG submitted an amendment to the topical report dated November 5, 1997, to change the trip devices STI from the proposed six months to a three month interval. The amendment listed the editorial changes to the topical report to reflect a three month STI and committed to incorporate these changes in the "approved" version of the topical report following staff approval. The staff finds this report, as amended, acceptable and agrees that the STI for the reactor trip devices can be extended for all B&W plants (except Three Mile Island) to the requested interval. Three Mile Island was not represented by the B&WOG on this issue.

In accordance with procedures established in NUREG-0390, "Topical Reports Review Status," we request that the B&WOG publish accepted revisions of BAW-10167, Supplement 3, within three months of receipt of this letter. The accepted versions should (1) incorporate all changes as per amendment, (2) incorporate this letter and the enclosed SER between the title page and the abstract, and (3) include an -A (designated accepted) following the report identification symbol.

Should our acceptance criteria or regulations change so that our conclusions as to the acceptability of this report are no longer valid, the B&WOG and/or the licensees referencing this topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued applicability of the topical report without revision.

Should you have any questions regarding the matters discussed above or the content of the enclosed SER, please contact I. Ahmed on (301) 415-3252.

Sincerely,

Thomas H. Essig

Thomas H. Essig, Acting Chief
Generic Issues and Environmental
Projects Branch
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Enclosure: As stated

cc: Mr. R. B. Borsum, Manager
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Rockville, MD 20852-1631

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

B&W OWNERS GROUP TOPICAL REPORT BAW-10167 SUPPLEMENT 3

JUSTIFICATION FOR INCREASING THE REACTOR TRIP SYSTEM ON-LINE TEST INTERVAL

1.0 INTRODUCTION

By letter dated June 7, 1996, (Reference 1), the B&W Owners Group (B&WOG) submitted Topical Report BAW-10167 Supplement 3, "Justification for Increasing the Reactor Trip System On-Line Test Intervals." This report was prepared by B&W Nuclear Technologies and provides the technical basis to justify increasing the on-line surveillance test interval (STI) from the current one-month to a six-month interval, for reactor trip system (RTS) trip devices consisting of reactor trip breakers (RTBs), reactor trip modules (RTMs), and electronic trip relays. By letters dated September 18, 1996 and January 3, 1997 (Reference 2), the B&WOG provided additional information to substantiate the topical report request. Subsequently, the B&WOG in their letter to the NRC dated November 5, 1997 (Reference 4), submitted an amendment to the topical report to change the requested STI for RTS trip devices from the proposed six months to a three month interval.

ENCLOSURE

The methodology and models used in this topical report are the same as those previously used in Supplement 1 of BAW-10167, "Justification for increasing the Reactor Trip System On-Line Test Intervals," which was submitted to justify the RTS instrument string STI extension from one-month to a six-month interval. At that time, the B&WOG chose not to include the RTS trip devices in their request for the STI extension because the RTB front-frame, active shunt trip, and lubricant upgrades (installed in response to Generic Letter (GL) 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events") were new, and it was prudent to test the trip devices more frequently until additional operating experience was obtained. The staff approved Supplement 1 of BAW-10167 and suggested some specific improvements in the methodology. Supplement 3 of BAW-10167 used an improved methodology and concentrated on the changes to the modeling and data that are necessary for examining the sensitivity of the RTS reliability to the RTS trip devices STI including updating of operating experience data. The reliability models used in this analysis are representative of both the Oconee (Oconee Units 1, 2 & 3, Crystal River Unit 3 and Arkansas Nuclear One Unit 1) and Davis Besse RTS design groups and do not include Three Mile Island which was not represented by the B&WOG on this issue. The unavailability of each of the two RTS trip device design groups is modeled in the report using reliability block diagrams for both the current one-month STI and the originally proposed six-month STI. The analysis evaluated the impact of the proposed STI extension on core melt frequency and RTS unavailability to demonstrate that the proposed STI change did not significantly increase plant risk when compared with the current technical specification requirements.

The following evaluation addresses both the acceptability of the probabilistic risk analysis presented in BAW-10167, Supplement 3 and the acceptability of the orginally proposed and amended STI extension.

2.0 EVALUATION

The staff evaluation included the following two aspects of the probabilistic risk analysis (PRA) performed by B&W Nuclear Technology to justify the proposed extension of the RTS trip devices STI:

(1) Models and data used for the reliability analysis

(2) Quantification of the analysis models

The methodology and models used in the BAW-10167, Supplement 3 analysis are the same as those used in the staff approved Supplement 1 of BAW-10167 including time-dependent, common mode failure and uncertainty analyses. Emphasis was placed on the use of operating experience for the data source in the derivation of both random and common mode failure rates. Nuclear Plant Reliability Data System (NPRDS), Licensee Event Reports (LER), a Sandia National Laboratory (SNL) research report, and the technical judgement of the maintenance technician or engineer were the sources for the RTS component failure database. Improvements were made in the Reliability Block Diagram (RBD) models of the RTS trip devices to incorporate increased details of the RTB failure data. Specifically, the RTB portions of the RBD were divided by failure mechanism into two components corresponding to the failures caused by cyclic (i.e., demand) stress and time-in-service-related (i.e., standby) stress. The RTB failure data reflected reliability improvements and reduction in the potential for common mode failures due to the implementation of the guidelines of GL 83-28.

In order to assess the sensitivity of RTS reliability to the trip device STI, the instrument string portion of the model was held constant (i.e., with six-month test intervals), and the testing frequency in the trip device portion of the model (RTMs, RTBs, and electronic trip relays) was varied from one-month to six-months. An error factor of 10 (the largest error factor listed in WASH-1400, "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, for instrumentation," and suggested in the staff safety evaluation report on Supplement 1 of BAW-10167) was used for the RTS trip devices random failure rate (lambda factor). When a common-mode failure rate could not be determined from the component failure history, a beta factor (fraction of lambda factor in which two or more components are involved due to common-mode failure) was used as suggested by NUREG/CR-5801, "Procedure for Analysis of Common-cause Failures in Probabilistic Safety Analysis." A beta factor of 0.05 was assumed in this analysis because the recent failure history of the RTS trip devices showed no evidence of multiple failures since the generic letter 83-28 upgrades were implemented.

A time-dependent and time-averaged RTS unavailability calculation was performed by B&W Nuclear Technologies using a reliability block diagram and computer codes for both the Oconee and Davis Besse RTS trip devices designs (Oconee design class plants use silicon control rectifiers (SCRs) to trip the regulating rods, groups 5 through 7 while the Davis Besse configuration uses the SCRs to trip both safety and regulating rods, groups 1 through 7). "All seven rod groups must trip" was used as the mission success criterion in the reliability model which made the quantification results more conservative (the most conservative success criterion for reactor trip used by the staff was defined in SECY 83-293 as insertion of half of the control rods into the core in a checkerboard pattern to shutdown the reactor). The quantification calculation also included plant spurious trip evaluation results which were directly attributed to surveillance testing of the RTS trip devices. This represented a net improvement in the number of scrams/plant/year and a reduction in core melt frequency (CMF) due to relaxation of the RTS trip devices STI.

Table 4-2 of BAW-10167, Supplement 3 (henceforth called Supplement 3), presents the failure rate data of the GE model AK RTBs used in B&W and Combustion Engineering designed plants. These data are primarily based on a NPRDS search (between 1988 and 1993 to update the data after implementation of GL 83-28 improvements) and on Sandia National Laboratory Report SAND-93-7027, "Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Switchgears". Figure 4-1 of Supplement 3 compares the failure rates of the RTBs before 1984 and after 1989. The number of RTB failures after 1989 is about one-sixth the failures before 1984. Considering that the results of the NPRDS search depends significantly on the search command used, the NRC staff conducted an independent search for this data base and came up with comparable results.

Using the Table 4-2 data of Supplement 3, the B&WOG calculated the RTS unavailabilities (failure/demand) for one month and six month test intervals using the computer program SAMPLE (Reference 3). The results of the computer run are presented in Table 6-1 and Figures 6-1 through 6-11 of Supplement 3. Figure 6-1 indicates that after 1989, the RTS availability went up seventeen times. This improvement in RTS availability was used to calculate the incremental risks of core damage frequency (CDF) from the extension of the RTS STI from one month to six months and the results are presented in Table 6-3. Table 6-3 indicates a net incremental risk in CDF of 3.2x10E-9 for Davis Besse and 2.6x10E-8 for Oconee type plants. From a risk significance point of view these values are acceptable. However, the staff could not directly verify the results without performing a computer analysis similar to the one performed for BAW-10167A, Supplement 1 by the Idaho National Laboratory. Therefore, the staff requested the B&WOG to provide an extrapolation of the data from BAW-10167A, Supplement 1 using a direct correlation methodology to show the reasonableness of the Supplement 3 results without reliance on computer analysis. The B&WOG's response in Reference 2 provided additional information including the following RTS failure probability per demand to establish the reasonableness of the Supplement 3 results:

DATA	DAVIS-BESSE GROUP	OCONEE GROUP
One-month test interval from Supplement 1	9E-9	1.1E-6
Six-month test interval extrapolated from Supplement 1	9.01E-9	2.6E-7
Six-month test interval from Supplement 3	9.05E-9	3.3E-7
Worst case sensitivity analysis of Supplement 3 data	9.12E-9	1.1E-6

The staff review of the information presented in References 1 and 2 indicates that Supplement 3 adequately demonstrated a negligible change in CMF and overall plant risk, and the conclusions drawn therein are reasonable. However, the staff determined that the availability of approximately five years of data collected between 1988 and 1993 indicating a six fold improvement in the number of RTB failures since implementation of GL 83-28 is not sufficient to offset the uncertainties associated with the actual potential breaker failure modes. As such, the staff did not find adequate operating history with test intervals longer than one month to support a change of STI from the current one month to six months. The B&WOG agreed with the staff concerns and submitted an amendment to Supplement 3 in Reference 4. The amendment requests a more conservative three month STI instead of the originally proposed six month interval.

The basis for techincal specification STI for any safety-related component or system is to ensure that the probability of an undetected failure existing within the component or system is small and to reduce the potential for spurious trips which may unnecessarily challenge the plant operators and safety systems. The B&WOG determined that the monthly test of the RTS trip devices was an unnecessary burden on utility resources because the reliability of the RTBs has considerably improved since implementation of the GL 83-28 recommendations and because there are sufficient data to confirm that the breaker upgrades were effective. Additionally, a three month STI has been approved by the staff for CE plants which use GE Model AK RTBs. Combustion Engineering Owners Group (CEOG) Topical Report CEN-327-A, "RPS/ESFAS Extended Test Interval Evaluation" dated January 1989 (Reference 5) which was approved by the staff included a three month STI for CE plant RTBs. Although the CEOG has not maintained a verifiable failure experience record of a three month STI for their RTBs, the B&WOG believes that a combined effort of the B&WOG and CEOG will increase the data pool for generating experience with the extended test interval. Nevertheless, in Reference 4 the B&WOG

identified that the laboratory tests performed by GE (manufacturers of the RTBs used at CE and B&W power plants) on the RTBs have shown that the upgraded lubricant will retain its stability well beyond 90 days. Further, the B&WOG has committed to monitor performance of the RTS trip devices to ensure that degradation does not occur as a result of the proposed STI extension. If performance criteria are exceeded, the B&WOG has committed to perform an evaluation and the feedback mechanism would alert the utilities to take corrective action, including more frequent testing.

3.0 CONCLUSION

Based on the above, the staff concludes that the analyses in BAW-10167, Supplement 3, as amended, adequately demonstrated a negligible change in CMF and overall plant risk, and thus the proposed extension of the RTS trip devices STI from the current one month to three months, is acceptable. Since the proposed change does not adversely impact plant safety, it is approved as an appropriate plant specific technical specification change for those B&W plant licensees covered by the BAW-10167, Supplement 3 analyses.

4.0 REFERENCES

1. Letter, J. H. Taylor to NRC Document Control Desk, dated June 7, 1996.

2. Letters, David J. Firth to J. L. Birmingham, dated, September 18, 1996 and January 3, 1997.

3. SAMPLE, General Purpose Computer Program for Uncertainty Analysis by Monte Carlo Simulation, original by WASH-1400 Reactor Safety Study Group, and modifications by E. Oelkers and T. L. Wilson, NPGD-TM-501, Rev. 1, Babcock & Wilson, Lynchburg, VA, April 1980.

4. Letter, M. W. Epling to J. L. Birmingham, dated November 5, 1997.

5. CEOG Topical Report CEN-327-A, "RPS/ESFAS Extended Test Interval Evaluation", dated January 1989.