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WESTINGHOUSE CLASS 3 (Non-Proprietary)

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Westinghouse Energy Systems

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WESTINGHOUSE CLASS 3 (Non-Proprietary)

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SUMMARY REPORT

PROCESS PROTECTION SYSTEM EAGLE 21 UPGRADE,

NSLB, MSS AND TTD IMPLEMENTATION

WATTS BAR UNITS 1 AND 2

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TABLE OF CONTENTS

TABLE OF CONTENTS
(continued)

LIST OF TABLES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

1.0 INTRODUCTION

Revision 1 incorporates TVA's comments and notes that TVA has decided not to insulate the Steam Generator Reference Legs, has performed a Feedline Break Analysis taking credit for a reactor trip being generated by High Containment Pressure instead of Low-Low Steam Generator Level (showing the Westinghouse analyses to be conservative), and has requested that the Westinghouse analyses and documentation continue to be based on taking credit for the Steam Generator Low-Low Level setpoint. Appendix A addresses the preceding with the intent of providing a clear definition of the boundaries and the "integration" or "overlaying' of the TVA analysis with the Westinghouse analysis that would be maintained from this day forward as long as the Steam Generator Reference Legs are not insulated.

The purpose of this Summary Report is to essentially describe the process protection system changes involved and the FSAR Chapter 15 Accidents safety analyses performed to support these changes for the Eagle 21 Process Protection System Upgrade, New Steamline Break (NSLB), Median Signal Selector (MSS) and Trip Time Delay (TTD).

The detailed plant changes are described in the Field Change Notices (FCNs) provided separately. The Safety Evaluation to support these changes (as described in this report and in the FCNs) and to provide the bases for the Setpoint Methodology Document (WCAP-12096, Revision 5) and the Technical Specification Mark-ups has been provided separately. The Technical Specification Mark-ups have also been provided separately. Additional supplemental documentation has been supplied separately, including the FSAR Chapters 7 and 15 Mark-ups, the Environmental And Seismic Qualification Test Report (WCAP-8687, Supplement 2-E69A, Revision 0), the Noise, Fault, Surge, And RFI Testing Report (WCAP-11733), and the Verification And Validation Report (WCAP-13191, Revision 2).

This section summarizes the reasons and benefits for implementing these changes, Section 2.0 describes all of the Chapter 15 Accidents requiring reanalysis to support these changes, including providing the analyses results, Section 3.0 describes the Process Protection System Changes, including the changes to the logic diagrams, Section 4.0 summarizes the effect of these changes on the Setpoint Methodology and the Technical Specifications, and Section 5.0 provides a listing of Supplemental Documents.

The Eagle 21 Process Protection System is a digital microprocessor based system which replaces the existing Foxboro analog system on a form, fit and functional basis. It provides the identical process protection functions as well as adding microprocessor capability. Features of Eagle 21 include the capability of automatic surveillance testing, self calibration, self diagnostics, and expansion for future upgrades. The protection system setpoints processed through the Eagle 21 racks were revised to reflect the different rack accuracies and elimination of the analog comparator. These Setpoint changes are listed in WCAP-12096, Revision 5. The Low Temperature Overpressure System (LTOPS) was addressed to account for the increase in instrument delay added by Eagle 21, which affects the PORV pressure setpoints. The Environmental and Seismic Qualification, Noise, Fault, Surge Withstand Capability, EMI And RFI Qualification Testing, and Verification And Validation documents are provided separately.

The New Steamline Break (NSLB) Protection System is being implemented to reduce the potential for spurious actuation of Safety Injection (SI) at low power by removing the requirement for coincident signals (for SI). High Steam Flow With Coincident Low-Low Tavg or Low Steamline Pressure signals are no longer used for SI actuation and steamline isolation. Also, SI actuation on High Differential Pressure has been deleted. These initiation signals have been replaced with SI actuation and steamline isolation on Low Steamline Pressure. A new steamline isolation signal actuated on High Negative

Steamline Pressure Rate has been added to provide protection when the Low Steamline Pressure signal is blocked in Mode 3. The applicable FSAR Chapter 15 accident (Major Rupture Of A Main Feedwater Pipe) has been reanalyzed. All later vintage Westinghouse plants have the New Steamline Break Protection System.

The Median Signal Selector (MSS) is being implemented to eliminate the low feedwater flow reactor trip to avoid the potential for spurious trips at low power levels during start-up. The low feedwater flow trip was previously used in conjunction with the steam generator low -low water level reactor trip to satisfy IEEE Standard 279-1971 control and protection system interaction. The MSS signal selector process eliminates the need for this interacting protection, as described in WCAP-12417. The Chapter 15 accidents do not require reanalysis because the low feedwater flow trip is not assumed to be the primary functioning reactor protection.

The implementation of the Trip Time Delay (TTD) reduces the potential for unnecessary steam generator low-low level reactor trips below 50% power by delaying the trip based upon the safety analysis acceptance criteria when the Steam Generator Low-Low Level setpoint is reached rather than initiating tripping at the time the setpoint is reached. The TTD varies with power level, with the longest delay at 0 % power. The delay time is based upon the power level and whether one steam generator or multiple (more than one) steam generators are involved. The digital Eagle 21 process protection system provides the capability of using a continuous time delay curve from 0 to 50 % power rather than requiring the selection of discrete power levels when using an analog system. The TTD methodology is described in WCAP-11325-P-A. The applicable FSAR Chapter 15 accidents (Full Power Loss of Normal Feedwater, Loss of Offsite Power to the Station Auxiliaries (Station Blackout), Major Rupture of a Main Feedwater Pipe, Steamline Break Mass/Energy Releases Outside Containment) have been reanalyzed.

2.0 NON-LOCA ANALYSES/EVALUATIONS

The following functional upgrades will be included with the Eagle 21 digital reactor protection scheduled to be installed in the Watts Bar plant before the startup of Units 1 and 2.

a) New Steamline Break Protection
b) Trip Time Delay (TTD)
c) Elimination of the Low Feedwater Flow Reactor Trip

This section addresses those non-LOCA safety analyses which were impacted by these changes to the protection system.

The digital electronics of the Eagle 21 system do not impact the non-LOCA safety analyses without the functional upgrades because the time delays and inaccuracies associated with the Eagle 21 system are not greater than those previously assumed for the analog system. The upgrades, however, affect the protection system modeling which was used in some of the original analyses of the licensing-basis non-LOCA transients.

Table 2.1 shows which non-LOCA analyses are affected by each of the upgrades such that reanalysis of the licensing-basis transient was required. The results of the analyses are presented after a brief discussion of the methodology used for each analysis. The reasoning behind why specific accidents are affected is provided in the following sections. It should be noted that the transients discussed in the following sections were analyzed because of specific upgrades. However, all the upgrades were incorporated where appropriate in each of the analyses.

TABLE 2.1

WATTS BAR UNITS 1 AND 2
NON-LOCA AFFECTED ACCIDENTS MATRIX

KEY: - = Evaluation X = Reanalyze

ACCIDENT	TTD	NEW SLB PROTECTION
ROD WITHDRAWAL FROM SUBCRITICAL	-	-
ROD WITHDRAWAL AT POWER	-	-
RCCA MISALIGNMENT	-	-
UNCONTROLLED BORON DILUTION	-	-
PARTIAL LOSS OF FLOW	-	-
STARTUP OF AN INACTIVE LOOP	-	-
LOSS OF LOAD/TURBINE TRIP	-	-
LOSS OF NORMAL FEEDWATER	X	-
FEEDWATER MALFUNCTION	-	-
EXCESSIVE LOAD INCREASE	-	-
RCS DEPRESSURIZATION	-	-
MSS DEPRESSURIZATION	-	-
INADVERTENT OPERATION OF ECCS	-	-
COMPLETE LOSS OF FLOW	-	-
MAIN STEAMLINE RUPTURE	-	-
MAIN FEEDWATER PIPE RUPTURE	X	-
SINGLE RCP LOCKED ROTOR	-	-
RCCA EJECTION	-	-
SLB M/E RELEASE INSIDE CONTAINMENT	-	-
SLB M/E RELEASE OUTSIDE CONTAINMENT	-	-
STEAMLINE BREAK AT POWER	-	-
STEAMLINE BREAK WITH RCCA WITHDRAWAL	-	-

2.1 New Steamline Break Protection

2.1.1 Functional Description

The current configuration of the Watts Bar Nuclear Plant reactor protection system includes safety injection and steamline isolation actuation logic commonly known in Westinghouse plants as Old Steamline Break Protection. With the introduction of the Eagle 21 digital electronics, the protection system will be upgraded to the most recent standard Westinghouse safety injection and steamline isolation actuation logic commonly known as New Steamline Break Protection. The differences between the logic are illustrated in Figure 2.1.1-1.

The New Steamline Break Protection System actuation of safety injection will result from any of the following.

1) Low Steamline Pressure
2) Low Pressurizer Pressure
3) High Containment Pressure

Steamline isolation is actuated by the following.

1) Low Steamline Pressure
2) High Steamline Pressure Rate
3) High-High Containment Pressure

The design basis non-LOCA safety analyses such as the Steamline Break Inside Containment, Accidental Depressurization of the Main Steam System, Inadvertent Operation of the Emergency Core Cooling System, Major Rupture of a Main Steamline, Major Rupture of a Main Feedline Pipe and Steamline Break Outside Containment specifically model steamline break protection functions although they may or may not actuate.

2.1.2 Impact of New Steamline Break Protection Logic on the Non-LOCA Safety Analyses

As part of the previous reanalysis of the Major Rupture of the Main Feedline Pipe and Steamline Break Outside Containment events for the Watts Bar Reduced Auxiliary Feedwater Program, New Steamline Break Protection logic was modeled for both events. In feedline break analyses for plants with Old Steamline Break Protection logic, safety injection does not occur because the setpoints are not reached. With the New Steamline Break Protection logic, however, the Low Steamline Pressure setpoint will be reached since the coincidence with High Steamline Flow has been eliminated. The potential impact of safety injection and steamline isolation on a feedline break event warrants the modeling of the logic during reanalysis. For the previous steamline break mass and energy release outside containment analysis with Old Steamline Break Protection logic, Low Steamline Pressure with coincident High Steam Flow signals actuated reactor trip, safety injection and steamline isolation for the three Double-Ended Rupture cases (see Reference 1). Thus, the reanalysis of the event included the logic for the New Steamline Break Protection System.

As documented in the Watts Bar Reduced Auxiliary Feedwater Safety Evaluation Checklist (Reference 2), acceptable results were obtained for the feedline break analysis assuming a minimum auxiliary feedwater rate of 410 gpm to two intact steam generators. Reference 2 also shows that twelve separate mass and energy release cases for the Steamline Break Outside Containment event were documented in Reference 3 for use by TVA in the evaluation of the effect of the implementation of the Reduced Auxiliary Feedwater and Eagle 21 programs on the environmental qualification of the equipment located outside containment.

The remaining events mentioned above are not impacted by the change from old steamline break protection to new steamline break protection. In these events, either Low Steamline Pressure (with coincident High Steam Flow), Low Pressurizer Pressure or High-High Containment Pressure signals actuated safety injection and steamline isolation. Of these signals, only the Low Steamline Pressure with coincident High Steam Flow is impacted by the logic change. The coincidence will be removed but the Low Steamline Pressure setpoint will remain unchanged. Although the coincidence logic will be removed, an evaluation has shown that the events would either benefit from an earlier actuation due to eliminating the coincidence or would not be impacted because Low Steamline Pressure was the second signal received.

2.2 Trip Time Delay (TTD)

The Steam Generator Low-Low Water Level Trip Time Delay (TTD) conceptual designs resulted from the Westinghouse Owners Group Trip Reduction and Assessment Program (WOG-TRAP) efforts to develop a means to reduce the frequency of unnecessary feedwater related reactor trips. The development of this concept is documented in WCAP-11325-P-A (Reference 4). In January 1981, the NRC issued Safety Evaluation Reports (SERs) approving the TTD conceptual designs of WCAP-11325-P-A for Westinghouse PWRs. As documented in the SERs, NRC approval is based on the review of a conceptual design for each system, representative functional requirements, description of the safety analysis methodology and generic safety analysis results. The SERs also list the licensing submittals that will be required by the NRC for review of plant-specific designs.

The Watts Bar Plant design is a Westinghouse digital implementation of the TTD logic located in each Steam Generator Low-Low Water Level protection set of the Eagle 21 Digital Protection system. This section is to provide safety analysis support, consistent with the requirements specified in the SERs, for the implementation of the TTD concepts in the Watts Bar Units. This section provides the following.

1. Basic functional description of the Watts Bar Plant TTD design

2. Results of calculations performed, consistent with the WCAP-11325-P-A approved methodology, to develop the Safety Analysis Limits (SALs) for the power dependent Steam Generator Low-Low Water Level Trip Time Delays

3. Evaluation of the impacts of the TTD SALs specified above on the non-LOCA safety analysis design basis

2.2.1 Trip Time Delay Functional Description

The conceptual design of WCAP-11325-P-A may be generally described as a system of pre-determined programmed trip delay times that are based upon (1) the prevailing power level at the time the Steam Generator Low-Low Water Level is reached and by (2) the number of generators that are affected. The TTD system is designed for low power or startup conditions so that once the Steam Generator Low-Low Water Level setpoint is reached, the TTD delays reactor trip and auxiliary feedwater actuation to allow time for operator corrective action or for natural stabilization of shrink/swell water level transients.

The Watts Bar TTD design is based on the introduction of a continuous time delay curve as a function of power (between 0 and 50% of rated thermal power), and the addition of a 2/4 steam generator trip logic to the existing 1/4 loop logic. The prevailing power level at the time the Steam Generator Low-Low Water Level setpoint is reached will be determined from the ΔT signal dedicated to the TTD logic. The protection system will enable the transmission of the Steam Generator Low-Low Water Level signal at the expiration of the enabled TTD delay if steam generator water level has not been recovered. Consistent with the WCAP-11325-P-A methodology, appropriate Safety Analysis Limits will therefore be determined for the following.

1/4 Steam Generator Logic
Indicated Power $\leq$ 50% of Rated Thermal Power (RTP)

Multiple Steam Generator Logic (at least 2/4 S/Gs)
Indicated Power $\leq$ 50% of RTP

No time delay is considered for this report for indicated power levels greater than 50% of RTP

When the Steam Generator Low-Low Water Level setpoint is reached, an elapsed trip time delay timer is actuated. As indicated above, the magnitude of the trip time delay is pre-set according to the power level with which it is interlocked and with the low-low logic path in which it is placed (i.e., low-low level in a single steam generator or low-low level in more than one steam generator).

If a low-low level condition is detected in one steam generator, then only the time delay that is associated with the single low-low level logic path and interlocked to the appropriate power level can satisfy the logic for transmission of the trip signal at the expiration of its trip delay. If at any time during this trip delay, a low-low level condition is detected in a second steam generator, then the time delay that is associated with the multiple low-low level logic path and interlocked to the appropriate power level can also satisfy the logic for transmission of the trip signal at the expiration of its trip delay. Since, at any given power level, the trip delay setpoint for two or more steam generators will be shorter than the trip delay setpoint for one low steam generator, reactor trip will occur at the end of the shorter effective trip delay, thus providing timely protective action for the more severe transient. Should the level be restored in all but one steam generator before the multiple affected steam generator time delay is expired, the remaining time from the single affected steam generator time delay will still be applied before reactor trip. Since the elapsed time trip delay timer is actuated by a single Steam Generator Low-Low Water Level signal, it is possible for a second steam generator to reach its low-low water level setpoint after the appropriate multiple low-low level trip delay has expired. In that case, the reactor trip signal would be transmitted without further delay.

If the power level decreases during a trip delay interval, this logic does not permit the lengthening of effective trip delays, which could result from switching to time delays associated with lower power ranges. If the power level increases, the effective trip delays are shortened as higher power levels become effective. If the water levels in all steam generators are not restored before the expiration of the shortest enabled trip delay, then the TTD logic transmits the Steam Generator Low-Low Water Level signal into the SSPS channel logic.

2.2.2 Trip Time Delay Safety Analysis Limit (SAL) Determination

Implementation of the TTD System in the Watts Bar Units 1 and 2 will require modification of the existing Steam Generator Low-Low Water Level protection system setpoints and the introduction of time delays. Consistent with the approved analysis methodology of WCAP-11325-P-A, analyses have been performed to determine revised SALs for input to the Steam Generator Low-Low Water Level Technical Specification limits. Watts Bar-specific Loss of Normal Feedwater and Major Rupture of a Main Feedline Pipe analyses have been performed to provide the Safety Analysis Limits for 1/4 and 2/4 logic time delay curves. The following cases were analyzed to determine Steam Generator Low-Low Water Level setpoint and time delay SALs.

- Loss of normal feedwater to four steam generators at 0%, 10%, 20%, 30%, 40% and 50% of RTP with a S/G Low-Low Level Setpoint = 0% NRS (Narrow Range Span)

- Loss of normal feedwater to one steam generator at 0%, 10%, 20%, 30%, 40% and 50% of RTP with a S/G Low-Low Level Setpoint = 0% NRS

- Feedline breaks to one steam generator at 10%, 20%, 30%, 40% and 50% of RTP with a S/G Low-Low Level Setpoint = 0% NRS

Initial time delay SALs for 1/N and 2/N logic were determined based on the Loss of Normal Feedwater analyses. These initial time delays were then implemented in the Feedline Break analyses. The results of the feedline break analysis showed that the initial time delay SALs for the 2/N logic had to be revised in order to obtain acceptable results. The final TTD SALs curves for both the low-low level signal in multiple steam generators and the low-low level signal in one steam generator are shown in Figures 2.2.2-1 and 2.2.2-2.

2.2.2.1 Part Power Loss of Normal Feedwater

Methods

The initial time delays were determined by permitting the results of the analysis to approach the acceptance criteria without permitting the part-power loss of normal feedwater analyses to be more limiting than the full power analysis discussed in Section 2.2.3.1. Since the Loss of Normal Feedwater event is an ANS Condition II transient, the criteria is that the Reactor Coolant System and Main Steam System pressure remain below 110% of system design pressure and the minimum DNBR remains above the safety analysis limit. In order to meet these requirements, the analysis should demonstrate that the auxiliary feedwater heat removal capacity is sufficient to offset the core decay heat and that the pressurizer does not fill.

The key analysis assumptions used for these cases are as follows:

1. Initial Conditions

   Consistent with the WCAP-11325-P-A analysis methodology, power level dependent initial conditions of 0%, 10%, 20%, 30%, 40% and 50% of RTP were assumed.

2. Decay Heat

   Consistent with the WCAP-11325-P-A analysis methodology, all cases used the ANS 1979 Decay Heat model (Reference 5).

3. Uncertainties

   Of particular importance to the part-power loss of normal feedwater cases is the uncertainty in power level indication since this function is integral to the TTD design. Each part-power case was analyzed assuming a maximum uncertainty in power level indication of 9% of RTP. For example, the analysis to determine the time delay at 10% of RTP had an initial power assumption of 19% of RTP.

4. Steam Generator Low-Low Water Level Setpoint

   The Steam Generator Low-Low Water Level setpoint assumed in these analyses is 0% NRS.

5. Steam Generator Low-Low Water Level Trip Time Delays

   The total Steam Generator Low-Low Water Level trip time delays assumed in the analysis of each part-power Loss of Normal Feedwater event include the initial SAL for the power level dependent time delays and an additional 2 second allowance for the time between receipt of the signal and when the control rods are free to drop.

6. 1/4 Loop Loss of Normal Feedwater

   The 1/4 loop Loss of Normal Feedwater cases assume a loss of normal feedwater to one steam generator. The loss of normal feedwater to one steam generator is not explicitly analyzed for the current Watts Bar FSAR since it is not necessary for any setpoint determination and its consequences, given the current plant automatic protection system, are bounded by those shown in the FSAR for loss of normal feedwater to all four steam generators. WCAP-11325-P-A introduced the analysis of loss of feedwater to one steam generator to support the concept of using 2/4 loop protection logic and 1/4 loop protection logic to respond to low level conditions in one or more steam generators.

7. Auxiliary Feedwater Flow Rate

   For the analysis of all the part power loss of normal feedwater events, failure of the Turbine Driven Auxiliary Feedwater pump is postulated as the most limiting single failure. As a result, auxiliary feedwater is assumed to be equally delivered to all four steam generators by the two motor driven auxiliary feedwater pump at a rate of 410 gpm. The auxiliary feedwater pump is assumed to start 60 seconds after the Steam Generator Low-Low Water Level setpoint is reached.

Results

A maximum acceptable time delay on the reactor trip and auxiliary feedwater actuation was calculated for each of the twelve cases discussed above. In each case, the auxiliary feedwater heat removal capability is sufficient to remove the decay heat and the pressurizer does not fill. These transient characteristics ensure that all applicable Condition II safety analysis acceptance criteria are met.

A representative sequence of events from the 20% of RTP Loss of Normal Feedwater to Four Steam Generators case and the 20% of RTP Loss of Normal Feedwater to One Steam Generator case is presented in Table 2.2.2.1-1 and 2.2.2.1-2. The transient response for these cases is shown in Figures 2.2.2.1-1 through 2.2.2.1-8.

2.2.2.2 Part Power Major Rupture of a Feedline Pipe

The Loss of Normal Feedwater analyses described in Section 2.2.2.1 established initial Safety Analysis Limits for the Steam Generator Low-Low Water Level signal delay times and trip setpoints. The purpose of the part-power feedline break analyses were to confirm that the final SALs permitted the results of the analysis to approach the acceptance criteria without permitting the part-power feedline break analyses to be more limiting than the full power analysis discussed in Section 2.2.3.2. The criteria for this ANS Condition IV event is that the core remains in place and geometrically intact with no loss of core cooling capability because the core remains covered with water. Pressurizer overfill and water relief is acceptable for this event.

The key analysis assumptions used for these cases are as follows:

1. Initial Conditions

   Power level dependent initial conditions of 10%, 20%, 30%, 40% and 50% of RTP were assumed.

2. Decay Heat

   Consistent with the WCAP-11325-P-A analysis methodology, all cases used the ANS 1979 Decay Heat model (Reference 5).

3. Uncertainties

   As with the part-power Loss of Normal Feedwater cases, the initial power level assumed a maximum uncertainty in power level indication of 9% of RTP.

4. Steam Generator Low-Low Water Level Setpoint

   The Steam Generator Low-Low Water Level setpoint assumed in these analyses is 0% NRS.

5. Steam Generator Low-Low Water Level Trip Time Delays

   The Steam Generator Low-Low Water Level trip time delays initially assumed in the analysis of each part-power Feedline Break case correspond to the time delays determined in the part-power Loss of Normal Feedwater analyses. Preliminary feedline break results, however, determined that the initial time delays for 2/N logic needed to be revised (lowered) in order to meet the safety analysis acceptable criteria.

6. Auxiliary Feedwater Flow Rate

   The auxiliary feedwater system is assumed to supply a total of 410 gpm to two unaffected steam generators as follows.

   a) The turbine driven pump is assumed to fail

   b) The motor driven pump supplying the faulted steam generator is assumed to conservatively spill all its flow out the break. The intact steam generator aligned to that pump is therefore assumed to receive no flow.

   c) The remaining motor driven pump supplies flow to two intact steam generators after a start delay of 60 seconds following reactor trip.

The part-power feedwater line break analysis was completed assuming both with and without available offsite power even though the without offsite power case is not considered to be credible because a reactor trip would occur upon loss of offsite power.

Results

In all the feedwater line rupture cases analyzed, the reactor protection system initiated the required protection functions in time to ensure the Condition IV acceptance criteria are met assuming the maximum 1/N logic time delays calculated in the part-power loss of normal feedwater analyses and 2/N logic time delays that were lower than the 2/N logic time delays calculated in the part-power loss of normal feedwater analyses.

Since the Trip Time Delay will be implemented as a continuous function of power, the data was fit to a third order polynomial equation. A separate curve was determined for both the low-low level signal in multiple steam generators and the low-low level signal in one steam generator. These curve are shown in Figures 2.2.2-1 and 2.2.2-2.

Figure 2.2.2.2-1 and Figure 2.2.2.2-2 show the calculated plant parameters following the feedline rupture transient for the worst case combination of break area and power level analyzed. This case is the 10% power double-ended rupture with offsite power. Figure 2.2.2.2-3 and Figure 2.2.2.2-4 shows the same case with loss of offsite power. The calculated sequences of events for both cases analyzed are presented in Table 2.2.2.2.

The major difference between the two 10% power cases can be seen in the plots of hot and cold leg temperatures. It is apparent from the initial portion of the transient that the case without offsite power results in higher temperatures in the hot leg. For longer times, however, the case with offsite power results in a more severe rise in temperature due to pump heat addition. Hence, water is relieved for the case with power. However, the core remains covered with water for both cases. The results also show that pressures in the RCS and Main Steam System remain below 110% of the respective design pressures.

All part-power feedline rupture transients are similar to the 10% power cases, except that as the initial power level increases and the break size decreases, the effective range for the Low Steamline Pressure safety injection reactor protection function decreases.

2.2.3 Impact on Non-LOCA Licensing Basis Safety Analyses

The analyses described in Sections 2.2.2.1 and 2.2.2.2 established Safety Analysis Limits for the Steam Generator Low-Low Water Level signal delay times and trip setpoint. The purpose of this section is to document the evaluation of these Safety Analysis Limits on the design basis safety analyses. A number of other events credit the Steam Generator Low-Low Water Level setpoint for reactor protection. These events must be analyzed with consistent assumptions on the Steam Generator Low-Low Water Level setpoint and in a manner which permits easy comparison of the results. As was previously stated, it was the intention of this project to keep, if possible, the current licensing basis events presented in the FSAR as the limiting transients. The transients which credit the Steam Generator Low-Low Water Level setpoint for reactor trip and auxiliary feedwater actuation include the following.

1) Full Power Loss of Normal Feedwater (FSAR Section 15.2.8)

2) Full Power Major Rupture of a Main Feedwater Pipe (FSAR Section 15.4.2.2)

3) Steamline Break Outside Containment (WCAP-13274)

The following sections will discuss the potential impact on each of the above licensing-basis events.

2.2.3.1 Full Power Loss of Normal Feedwater (FSAR Section 15.2.8)

Loss of Normal Feedwater is analyzed for the Watts Bar Plant in FSAR Section 15.2.8. This Condition II accident postulates a loss of normal feedwater to all four steam generators. The FSAR loss of normal feedwater analysis is performed to demonstrate the adequacy of the reactor protection system and engineered safeguards systems (i.e., the auxiliary feedwater system) in removing long-term decay heat and preventing excessive heatup of the RCS with possible resultant overpressurization or loss of RCS water inventory. The FSAR safety analysis assumptions are conservatively chosen to maximize the resulting primary side heat-up transient and, therefore, the dependency on the auxiliary feedwater system to adequately remove decay heat. The FSAR transient assumes full power initial conditions and accident protection from a receipt of a Steam Generator Low-Low Water Level signal.

The current licensing-basis Loss of Normal Feedwater safety analysis was performed as part of the Watts Bar Reduced Auxiliary Feedwater Program. As documented in Reference 2, the transient results indicate that all the Condition II acceptance criteria were met assuming an auxiliary feedwater flowrate of 820 gpm equally delivered to all four steam generators. Given that the reanalysis of the Loss of Normal Feedwater event assumed a Steam Generator Low-Low Water Level setpoint of 0% Narrow Range Span, the ANS 1979 Decay Heat model and initial condition and protection system uncertainties and allowances corresponding to the Eagle 21 program, the conclusions remain valid for the TTD SALs determined in Sections 2.2.2.1 and 2.2.2.2.

2.2.3.2 Full Power Major Rupture of a Main Feedline Pipe (FSAR Section 15.4.2.2)

A Reactor Coolant System heatup caused by a main feedwater line rupture is a Condition IV transient analyzed for the Watts Bar plant in FSAR Chapter 15.4.2.2. This transient was discussed in Section 2.1.2 in relation to the change from old steamline break protection to new steamline break protection. Results of the Feedline break transient, with and without offsite power, are presented in the FSAR to assure that the overheating, and consequently the radiation release limits of 10CFR100 are not exceeded. The FSAR transients are performed assuming full power initial conditions. For the present protection system, this assumption maximizes the heatup. Acceptable transient results demonstrate that:

i. Peak transient Reactor Coolant System and Main Steam System pressures are less than 110% of design pressures

ii. Sufficient liquid in the Reactor Coolant System is maintained so that the core remains in place and geometrically intact with no loss of core cooling capability. This criterion is met by ensuring that bulk boiling does not occur before the transient is turned around by auxiliary feedwater addition

The feedline break transient presented in the Watts Bar FSAR assumes reactor trip and actuation of auxiliary feedwater to occur due to receipt of a Steam Generator Low-Low Water Level signal. Each of these safety features actuations is essential for the successful mitigation of the accident consequences as conservatively predicted by the safety analyses. Rod insertion due to automatic reactor trip terminates the nuclear power contribution to the primary heatup. The delivery of auxiliary feedwater is essential for the removal of core decay heat and, therefore, the prevention of fuel damage and core uncovery.

The current licensing-basis Feedline Break safety analysis was performed as part of the Watts Bar Reduced Auxiliary Feedwater Program. As documented in Reference 2, the transient results indicate that all the acceptance criteria listed above were met assuming an auxiliary feedwater flowrate of 410 gpm equally delivered to two intact steam generators. Given that the reanalysis of the feedline break event assumed a Steam Generator Low-Low Water Level setpoint of 0% Narrow Range Span, the ANS 1979 Decay Heat model, new steamline break protection and initial condition and protection system uncertainties and allowances corresponding to the Eagle 21 program, the conclusions remain valid for the TTD SALs determined in Sections 2.2.2.1 and 2.2.2.2.

2.2.3.3 Steamline Break Outside Containment

The Westinghouse Steamline Break Mass and Energy Releases Outside Containment analysis, as documented in Reference 2, was reanalyzed as part of the Watts Bar Reduced Auxiliary Feedwater Program. The resulting mass and energy releases for the twelve cases analyzed are documented in Reference 3. The analysis, with the knowledge that the Watts Bar Plants would be implementing the Eagle 21 functional upgrades, assumed new steamline break protection, the ANS 1979 decay heat model, a Steam Generator Low-Low Water Level Safety Analysis setpoint of 0% NRS and initial condition and protection system uncertainties and allowances corresponding to the Eagle 21 program. The power levels examined for this event include 30%, 70%, and 100% of RTP. Since the implementation of the TTD System in the Watts Bar Units introduces time delays at indicated power levels lower than 50% of RTP, four Main Steam Line Break Mass and Energy Releases Outside Containment cases are potentially impacted by the TTD upgrade since these cases were analyzed at 30% of rated thermal power and credited reactor trip and auxiliary feedwater actuation on a Steam Generator Low-Low Water Level signal without a TTD. An evaluation was performed for the four cases to determine if the predicted mass and energy releases for these cases would be affected by the TTD SALs on the Steam Generator Low-Low Water Level setpoint at 30% power. The results of the evaluation showed that the mass and energy releases for the four 30% power cases would not be impacted by the modeling of the TTD system. Therefore, it is concluded that introduction of the TTD system will not invalidate the conclusions presented in References 2 and 3 or in Section 2.1.2 of this report.

2.2.4 TTD/AMSAC Interaction

The Trip Time Delay system consists of pre-determined programmed steam generator low-low water level reactor trip and auxiliary feedwater delay times that are based upon the highest power level observed during a steam generator low-low water level transient at the time the Steam Generator Low-Low Water Level setpoint is reached and the number of steam generators that are affected. In the Watts Bar design, the trip delay times are determined from the TTD Safety Analysis Limit curves for one and multiple steam generators as a function of power (below 50% of RTP).

The design of the ATWS Mitigation System Actuation Circuitry (AMSAC), as described in FSAR Section 7.7.1.12, provides an independent backup to the existing reactor protection system to initiate a turbine trip and actuates auxiliary feedwater flow in the event of an anticipated transient without a reactor trip while the power level is above 40% of RTP. AMSAC is required by 10CFR50.62. In the Watts Bar units, AMSAC will trip the turbine and start the auxiliary feedwater pumps if the water level in three or four steam generators drops below the AMSAC setpoint (which is set at 5% NRS below the steam generator low-low level trip setpoint) and the power level is 40% of RTP or greater. The AMSAC functions are delayed at Watts Bar by 30 seconds.

In the 40% to 50% power range, if the water levels in three or four steam generators drop below the AMSAC setpoint, the AMSAC and TTD systems will both be actuated. In the affected power range, the time delays from the TTD SAL curve multiple steam generators are shorter than the AMSAC delay time. With this, the TTD system will initiate reactor trip, startup of the Auxiliary Feedwater pumps and turbine trip on reactor trip before the AMSAC system can initiate turbine trip and startup of the Auxiliary Feedwater pumps. Therefore, there is no potential for the AMSAC system to interact with the TTD system.

2.3 Elimination of the Low Feedwater Flow Reactor Trip

Elimination of the Low Feedwater Flow Reactor trip does not require any reanalysis of the non-LOCA safety analyses because this trip was never assumed to be the primary functioning reactor protection. However, for a plant which only has three steam generator level transmitters per channel for Steam Generator Low-Low Water Level protection, these transmitters also provide signals used for control (i.e., feedwater flow control). As a result, the low feedwater flow reactor trip function was implicitly credited as a diverse trip to the Steam Generator Low-Low Water Level reactor trip function to address control/protection interaction concerns.

The elimination of the Low Feedwater Flow reactor trip function for Watts Bar will be accomplished by the introduction of a median signal selector (MSS). The addition of a MSS addresses any control and protection interaction concerns and insures that the removal of the subject trip does not impact (explicitly or implicitly) the non-LOCA safety analyses.

2.4 Conclusion

The preceding discussion demonstrated that the functional upgrades associated with the introduction of the Eagle 21 digital protection system are acceptable with respect to the non-LOCA safety analyses. The analysis results met all applicable safety criteria.

2.5 Non-LOCA References

1. WCAP-11053 (W Proprietary Class 2), "Steamline Break Outside Containment Mass and Energy Release Analysis (TVA - Watts Bar)," March 29, 1985.

2. Letter WAT-D-8935, "TENNESSEE VALLEY AUTHORITY WATTS BAR NUCLEAR PLANT UNIT NUMBERS 1 and 2, Auxiliary Feedwater Flow Program Safety Evaluation (SECL-92-120)," July 31, 1992.

3. WCAP-13274 (W Proprietary Class 2), "Steamline Break Outside Containment Mass/Energy Release for the Eagle 21/Reduced Auxiliary Feedwater Flow Programs (TVA - Watts Bar)," April 1992.

4. WCAP-11325-P-A Rev. 1, "Steam Generator Low Water Level Protection System Modifications to Reduced Feedwater-Related Trips," February 1988.

5. ANSI/ANS-5.1-1979, American National Standard for Decay Heat Power in Light Water Reactors, August 29, 1979.

TABLE 2.2.2.1-1

TIME SEQUENCE OF EVENTS

LOSS OF NORMAL FEEDWATER TO FOUR STEAM GENERATORS
AT 20% RATED THERMAL POWER

EVENT	TIME (sec)
Main feedwater flow stops	10.0
Steam Generator Low-Low Water Level setpoint reached in multiple Steam Generators	184.6
Steam Generator Low-Low Water Level signal transmitted	289.6
Rods begin to drop	291.6
Maximum water level in pressurizer occurs	295.0
Two motor driven auxiliary feedwater pumps start	349.6

TABLE 2.2.2.1-2

TIME SEQUENCE OF EVENTS

LOSS OF NORMAL FEEDWATER TO ONE STEAM GENERATORS
AT 20% RATED THERMAL POWER

EVENT	TIME (sec)
Main feedwater flow stops to one steam generator	10.0
Steam Generator Low-Low Water Level setpoint reached in faulted Steam Generator	173.5
Steam Generator Low-Low Water Level signal transmitted	303.5
Rods begin to drop	305.5
Maximum water level in pressurizer occurs	308.0
Two motor driven auxiliary feedwater pumps start	363.5

TABLE 2.2.2.2

TIME SEQUENCE OF EVENTS

MAJOR RUPTURE OF A MAIN FEEDWATER PIPE
AT 10% RATED THERMAL POWER

CASE	EVENT	TIME (sec)
With Offsite Power Available	Main Feedline rupture occurs	10.0
	Steam Generator Low-Low Water Level setpoint reached in faulted Steam Generator	37.5
	Steam Generator Low-Low Water Level signal transmitted	229.0
	Rods begin to drop	231.0
	One motor driven auxiliary feedwater pump begin providing flow to the two unaffected intact Steam Generators	289.0
	Low steamline pressure setpoint reached	394.9
	Pressurizer water relief begins	1616
	Core decay heat plus pump heat decreases to auxiliary feedwater heat removal capacity	4488

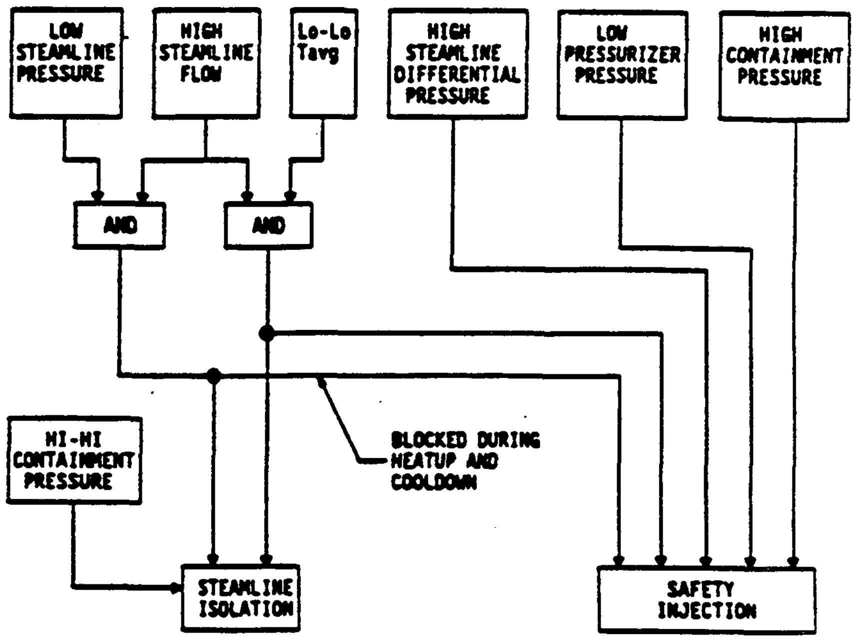
TABLE 2.2.2.2

TIME SEQUENCE OF EVENTS

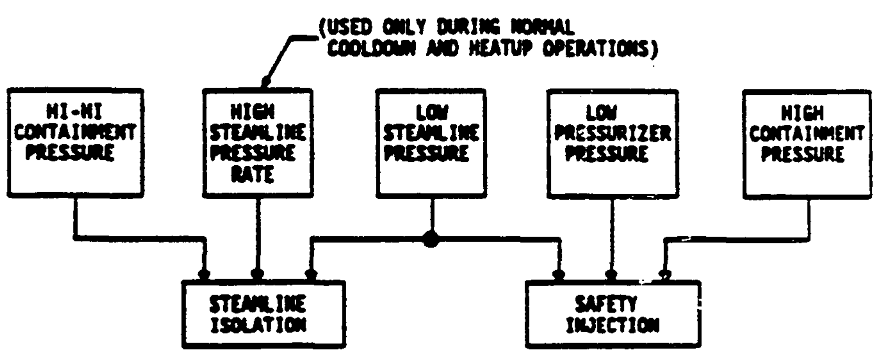
MAJOR RUPTURE OF A MAIN FEEDWATER PIPE
AT 10% RATED THERMAL POWER

(Continued)

CASE	EVENT	TIME (sec)
Without Offsite Power Available	Main Feedline rupture occurs	10.0
	Steam Generator Low-Low Water Level setpoint reached in faulted Steam Generator	37.5
	Steam Generator Low-Low Water Level signal transmitted	229.0
	Rods begin to drop	231.0
	One motor driven auxiliary feedwater pump begin providing flow to the two unaffected intact Steam Generators	289.0
	Low steamline pressure setpoint reached	404.7
	Core decay heat decreases to auxiliary feedwater heat removal capacity	1446
	Pressurizer water relief begins	4816



SYSTEM 1 (OLD STEAMLINE BREAK PROTECTION)



SYSTEM 2 (NEW STEAMLINE BREAK PROTECTION)

FIGURE 2.1.1-1

Steamline Break Protection Systems

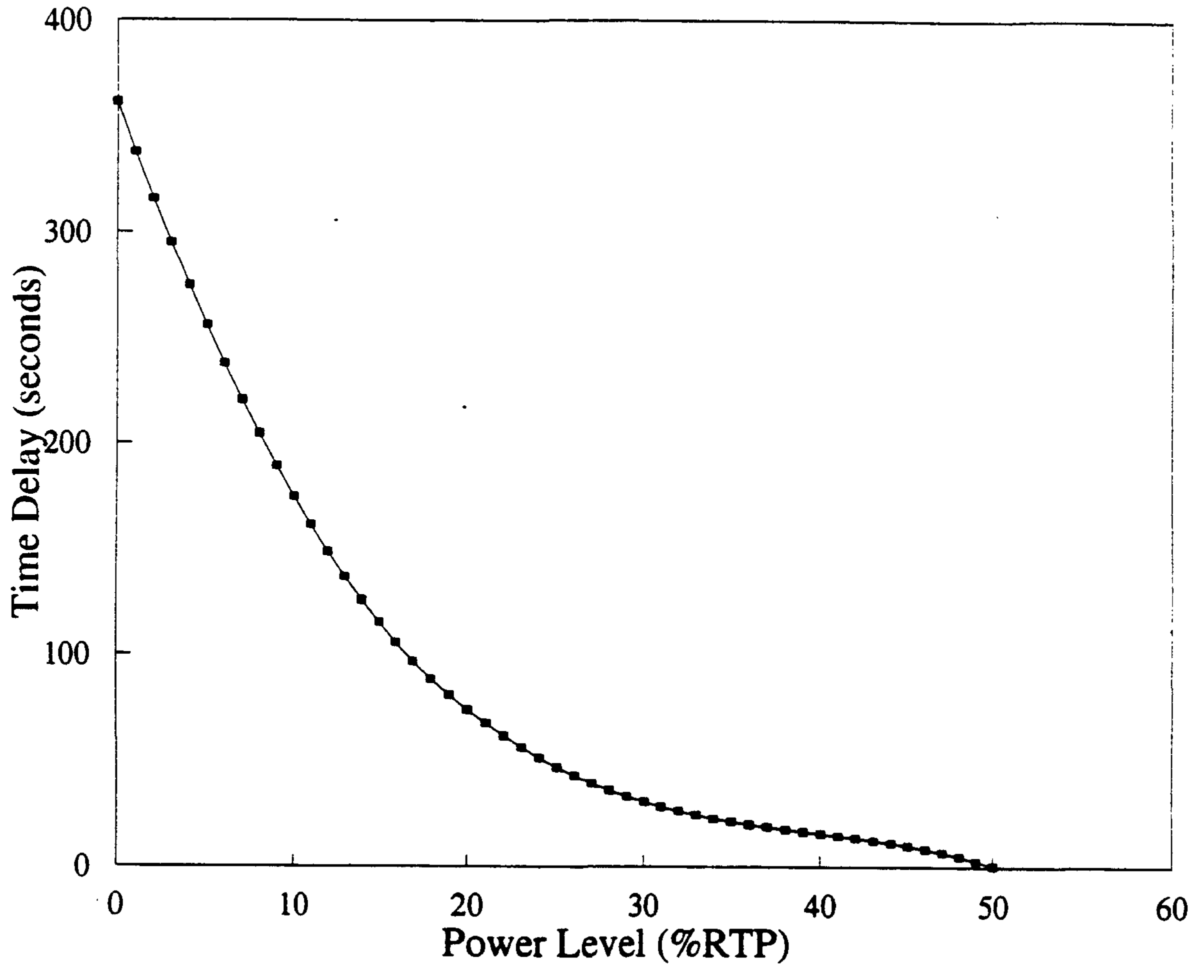


FIGURE 2.2.2-1

Calculated Trip Time Delays versus Power for Steam Generator Low-Low Water Level in Multiple Steam Generators

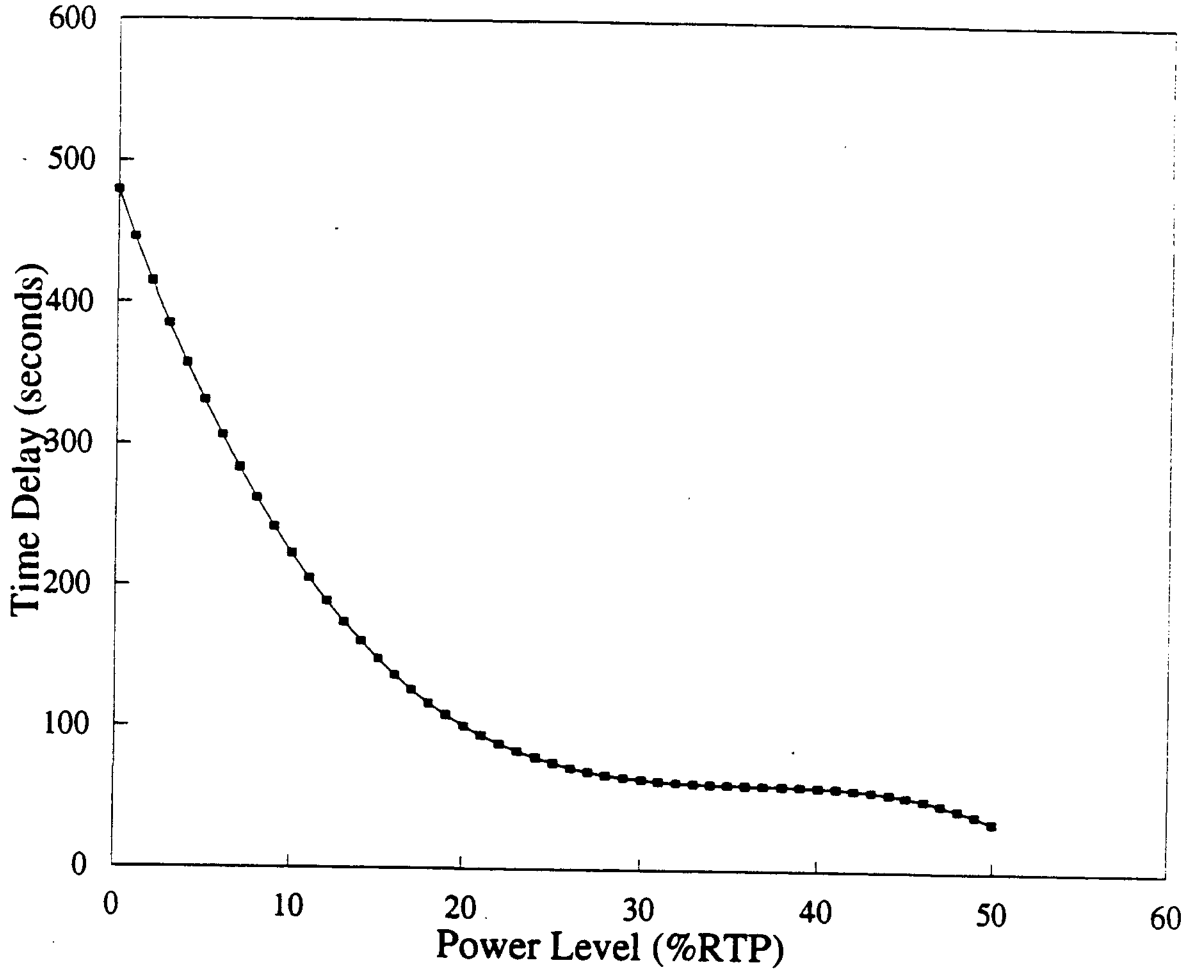
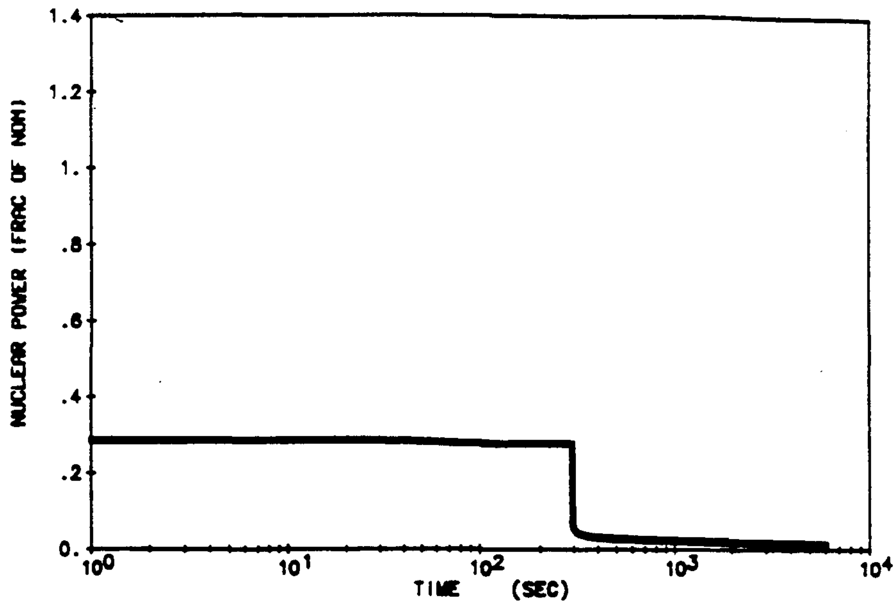


FIGURE 2.2.2-2

Calculated Trip Time Delays versus Power for Steam Generator Low-Low Water Level in One Steam Generator



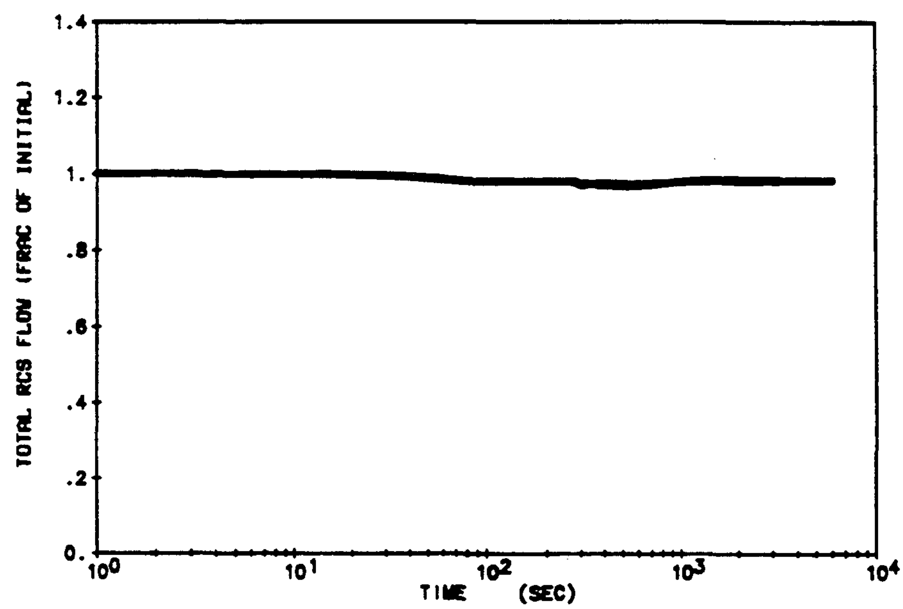


FIGURE 2.2.2.1-1

PART-POWER LOSS OF NORMAL FEEDWATER TO FOUR STEAM GENERATORS - 20% RTP with N/4 Trip Time Delay - Nuclear Power and RCS Flow versus Time

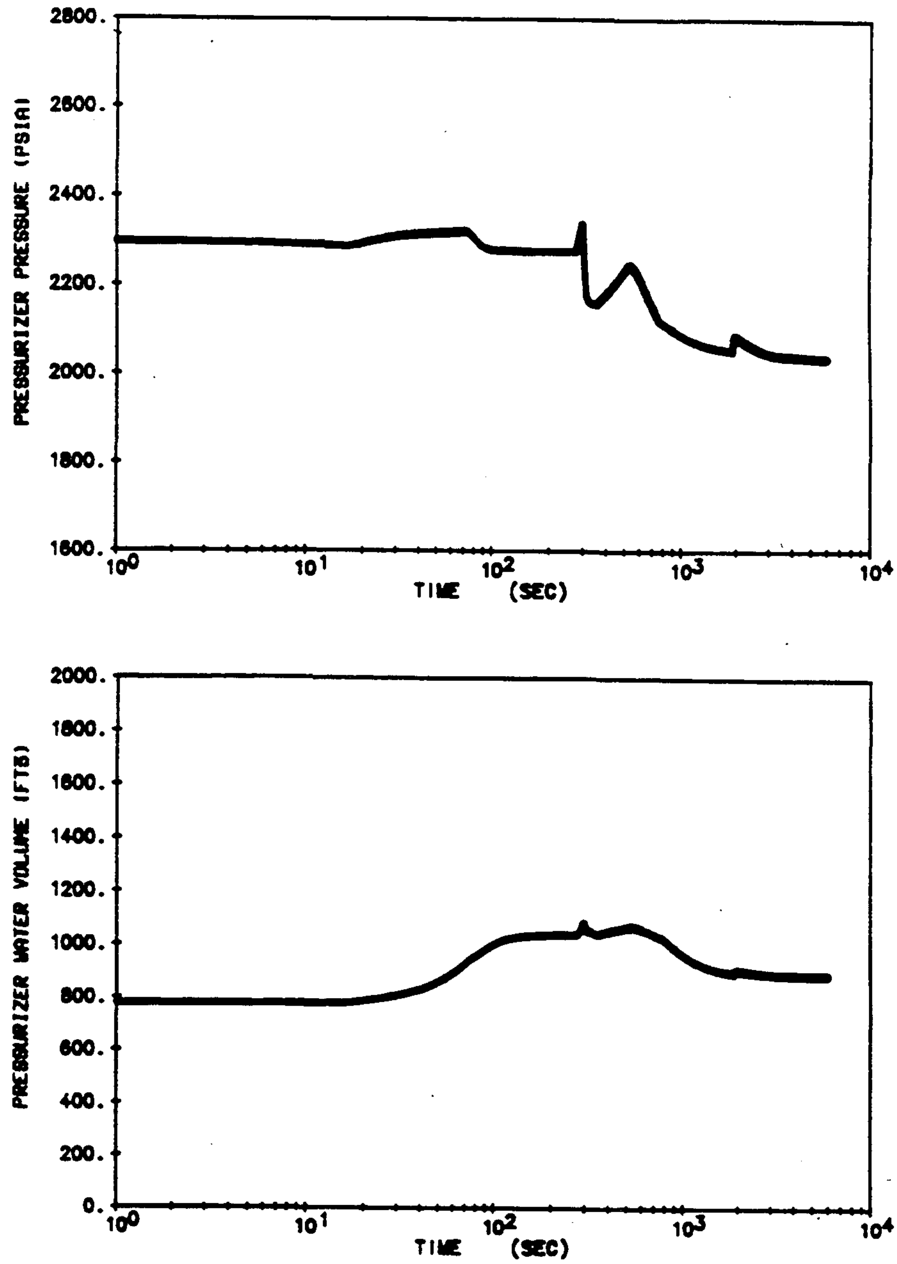


FIGURE 2.2.2.1-2

PART-POWER LOSS OF NORMAL FEEDWATER TO FOUR STEAM GENERATORS - 20% RTP with N/4 Trip Time Delay - Pressurizer Pressure and Pressurizer Water Volume versus Time

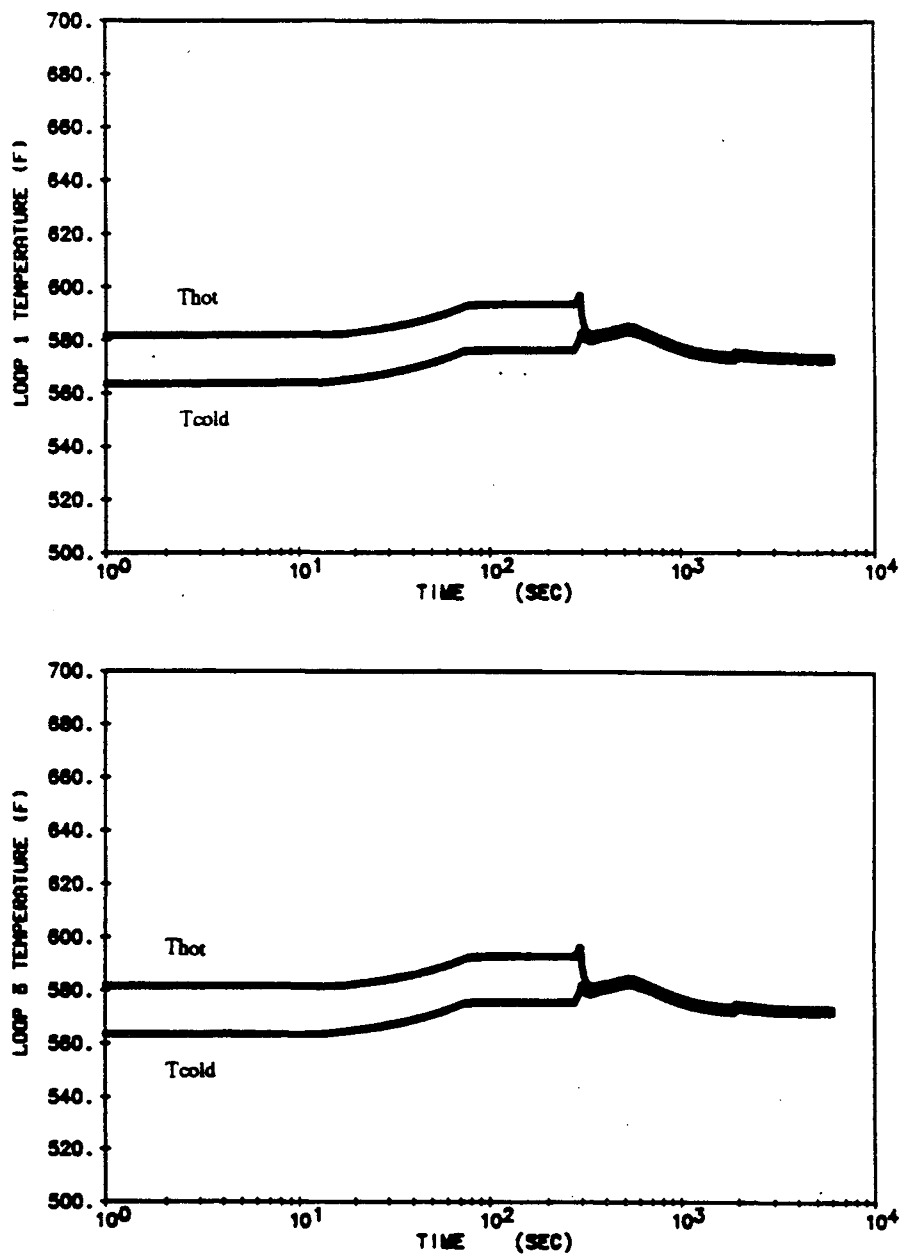


FIGURE 2.2.2.1-3

PART-POWER LOSS OF NORMAL FEEDWATER TO FOUR STEAM GENERATORS - 20% RTP with N/4 Trip Time Delay - Loop 1 and 3 Cold Leg and Hot Leg Temperatures versus Time

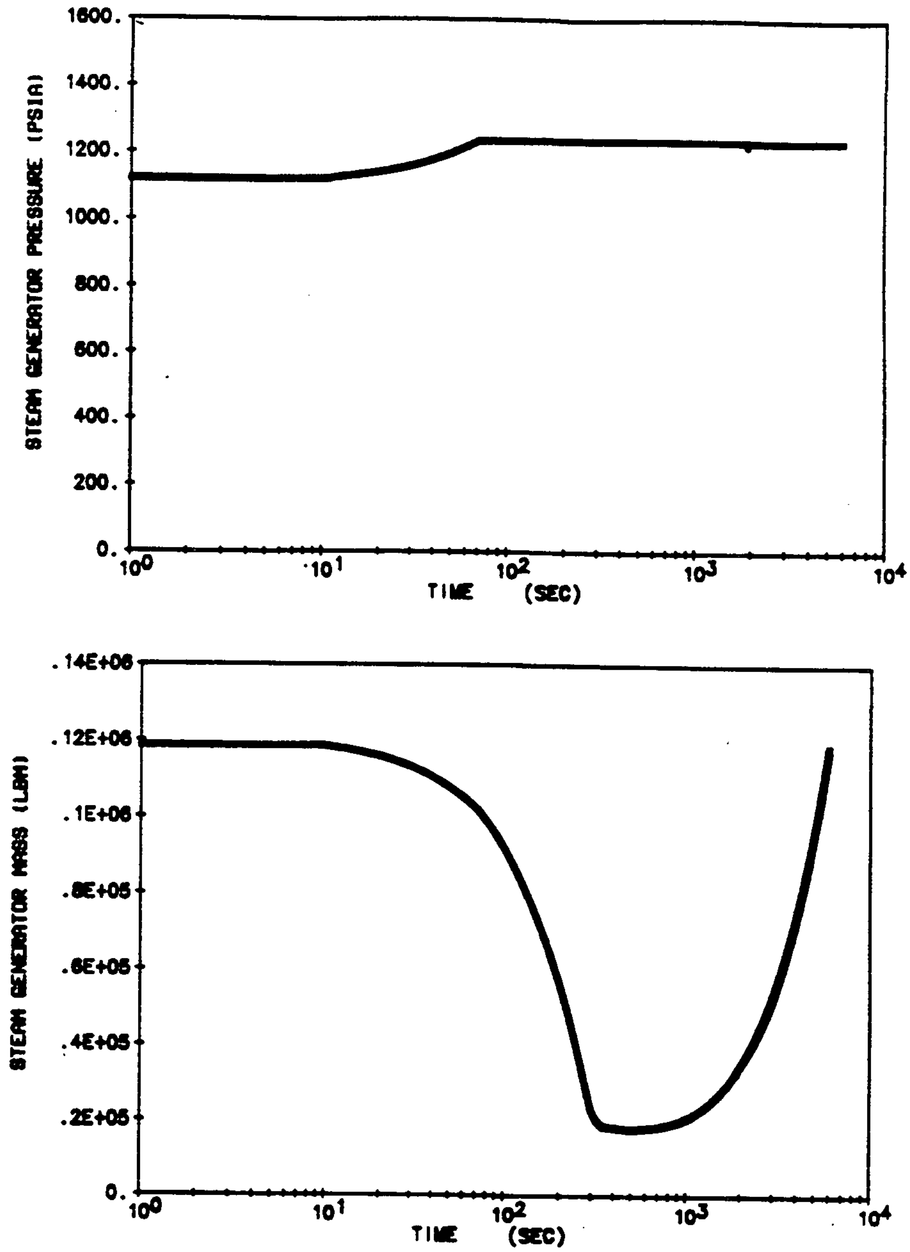


FIGURE 2.2.2.1-4

PART-POWER LOSS OF NORMAL FEEDWATER TO FOUR STEAM GENERATORS - 20% RTP with N/4 Trip Time Delay - Steam Generator Pressure and Steam Generator Mass versus Time

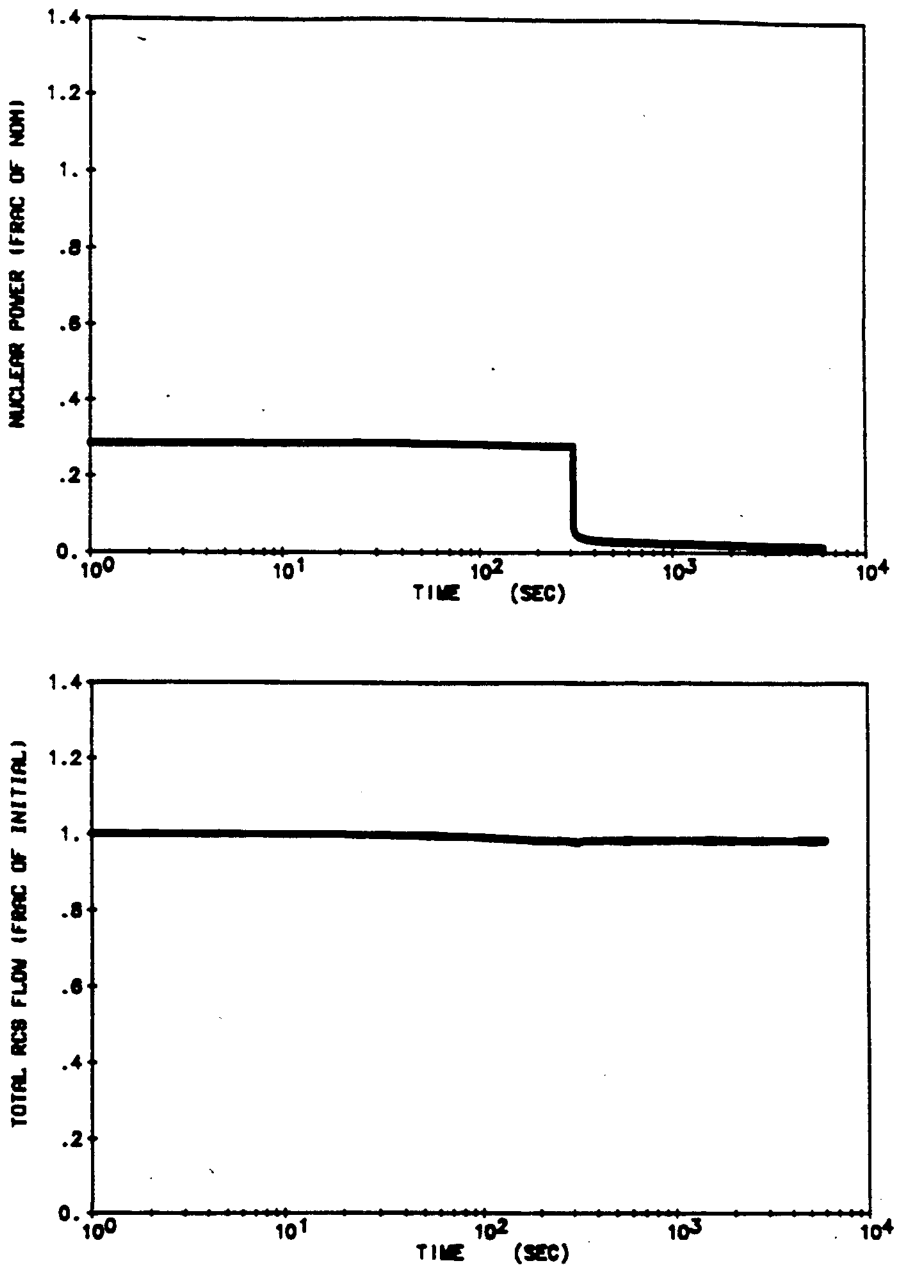
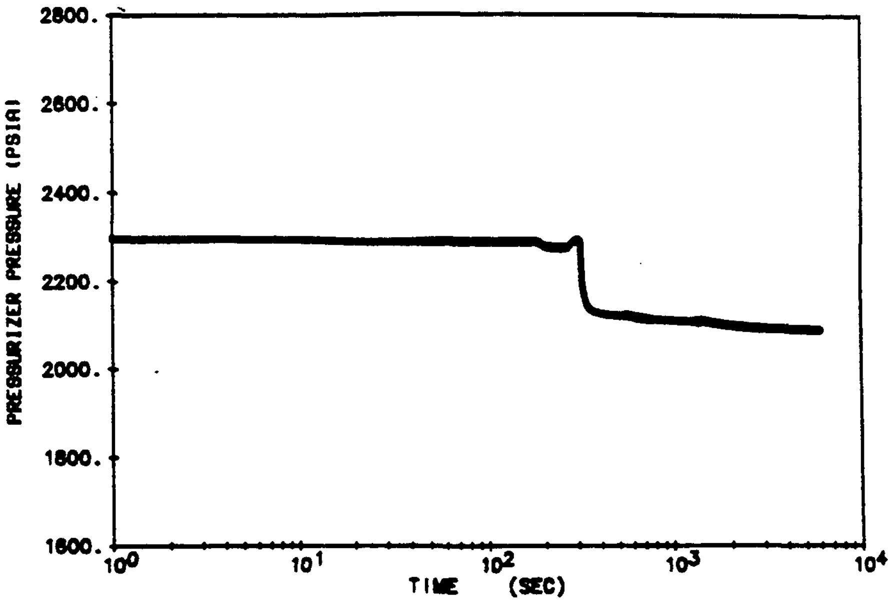


FIGURE 2.2.2.1-5

PART-POWER LOSS OF NORMAL FEEDWATER TO ONE STEAM GENERATOR - 20% RTP with 1/4 Trip Time Delay - Nuclear Power and RCS Flow versus Time



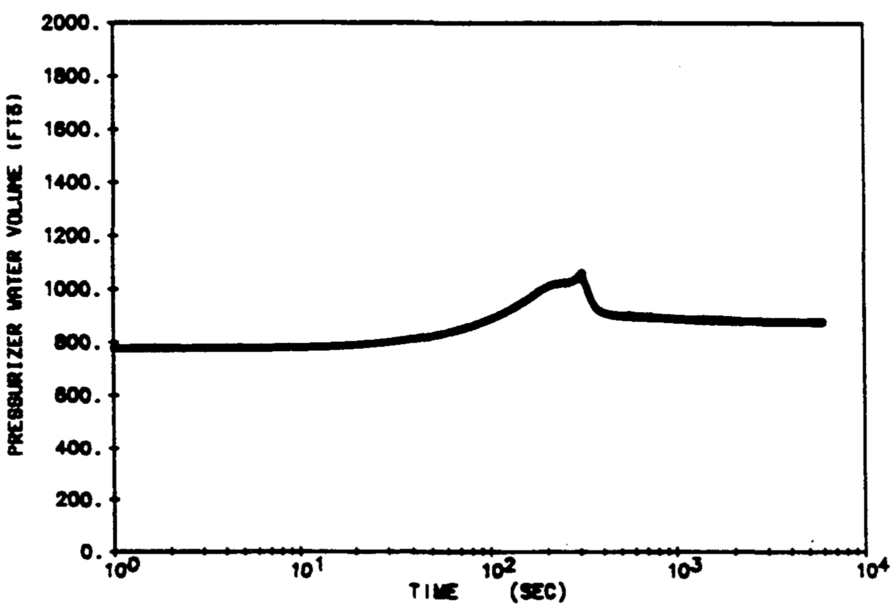


FIGURE 2.2.2.1-6

PART-POWER LOSS OF NORMAL FEEDWATER TO ONE STEAM GENERATOR - 20% RTP with 1/4 Trip Time Delay - Pressurizer Pressure and Pressurizer Water Volume versus Time

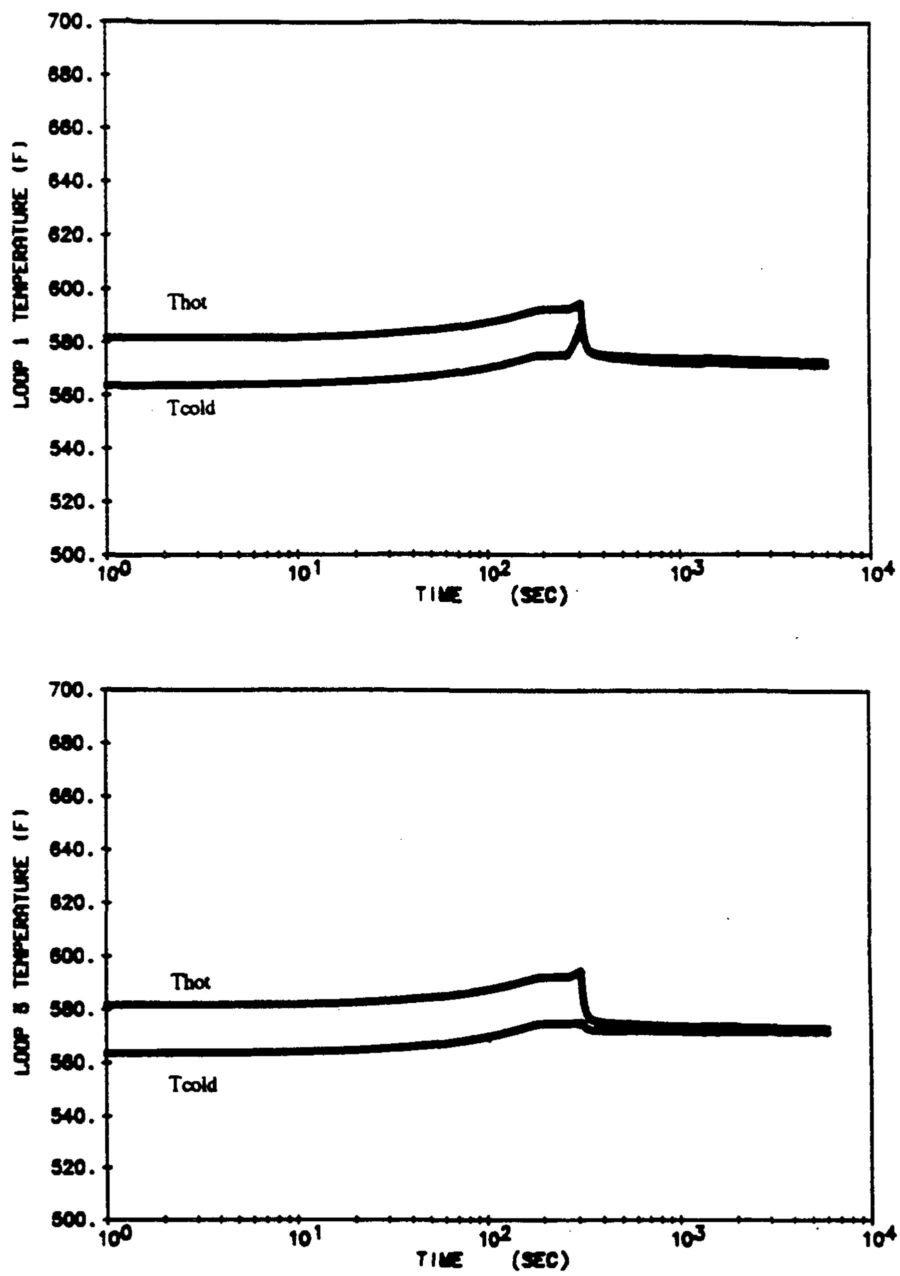
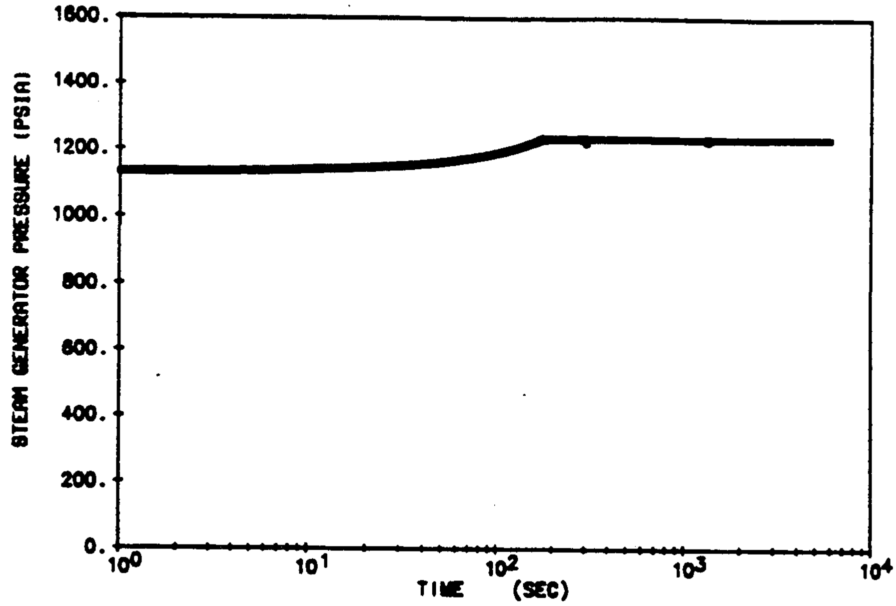


FIGURE 2.2.2.1-7

PART-POWER LOSS OF NORMAL FEEDWATER TO ONE STEAM GENERATOR - 20% RTP with 1/4 Trip Time Delay - Loop 1 and 3 Cold Leg and Hot Leg Temperatures versus Time



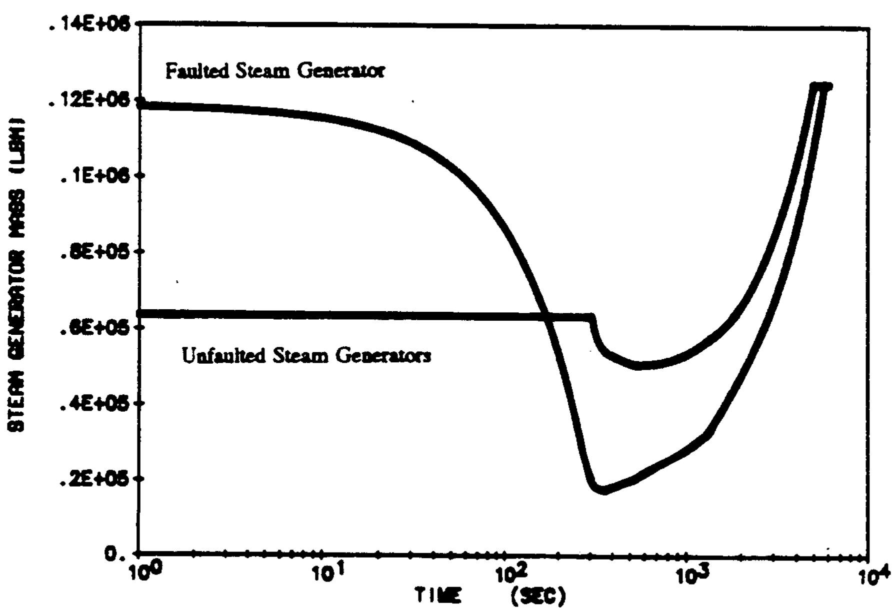


FIGURE 2.2.2.1-8

PART-POWER LOSS OF NORMAL FEEDWATER TO ONE STEAM GENERATOR - 20% RTP with 1/4 Trip Time Delay - Steam Generator Pressure and Steam Generator Mass versus Time

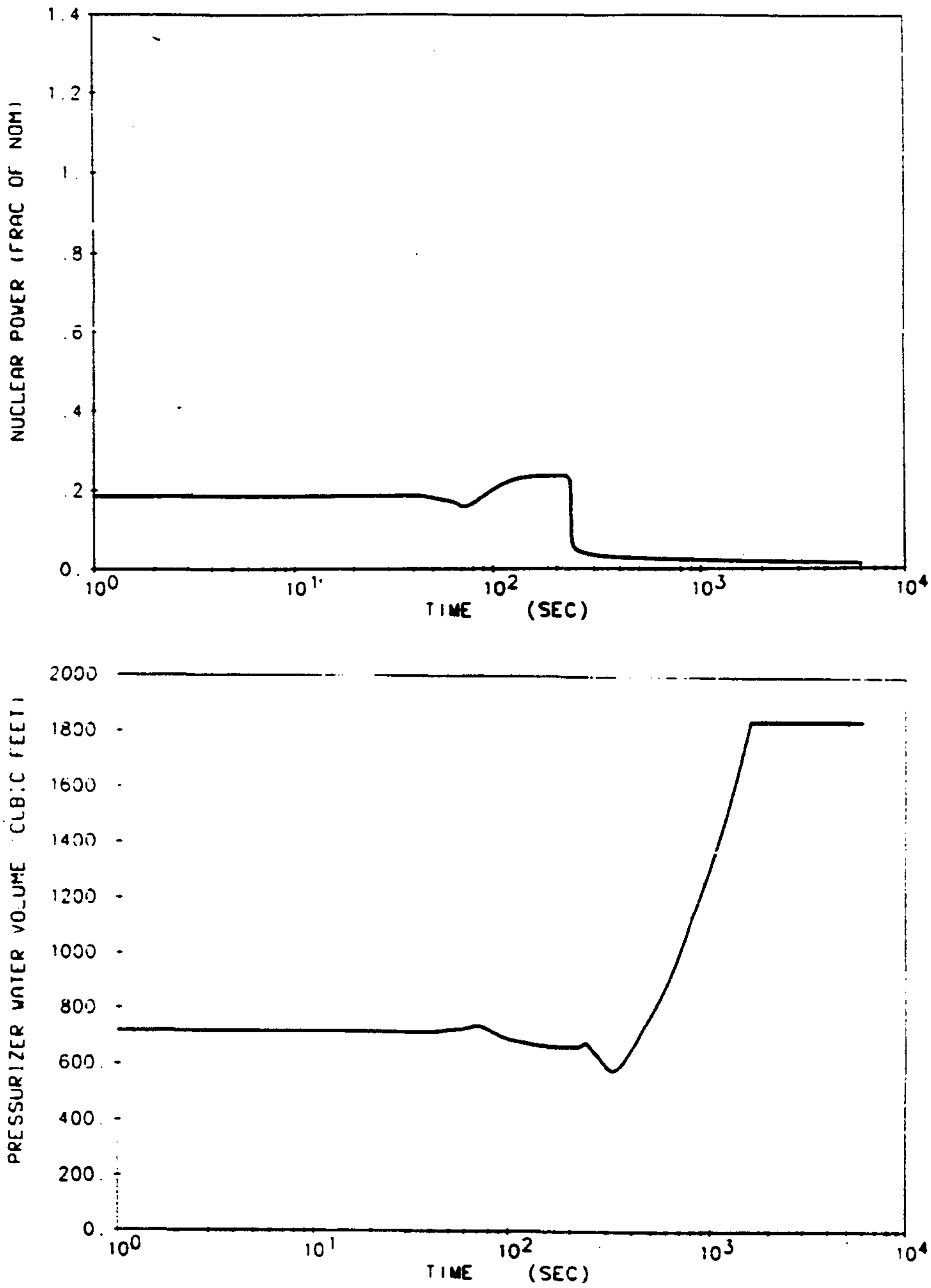


FIGURE 2.2.2.2-1

MAJOR RUPTURE OF A MAIN FEEDWATER PIPE AT 10 PERCENT POWER, WITH OFFSITE POWER - Nuclear Power and Pressurizer Water Volume versus Time

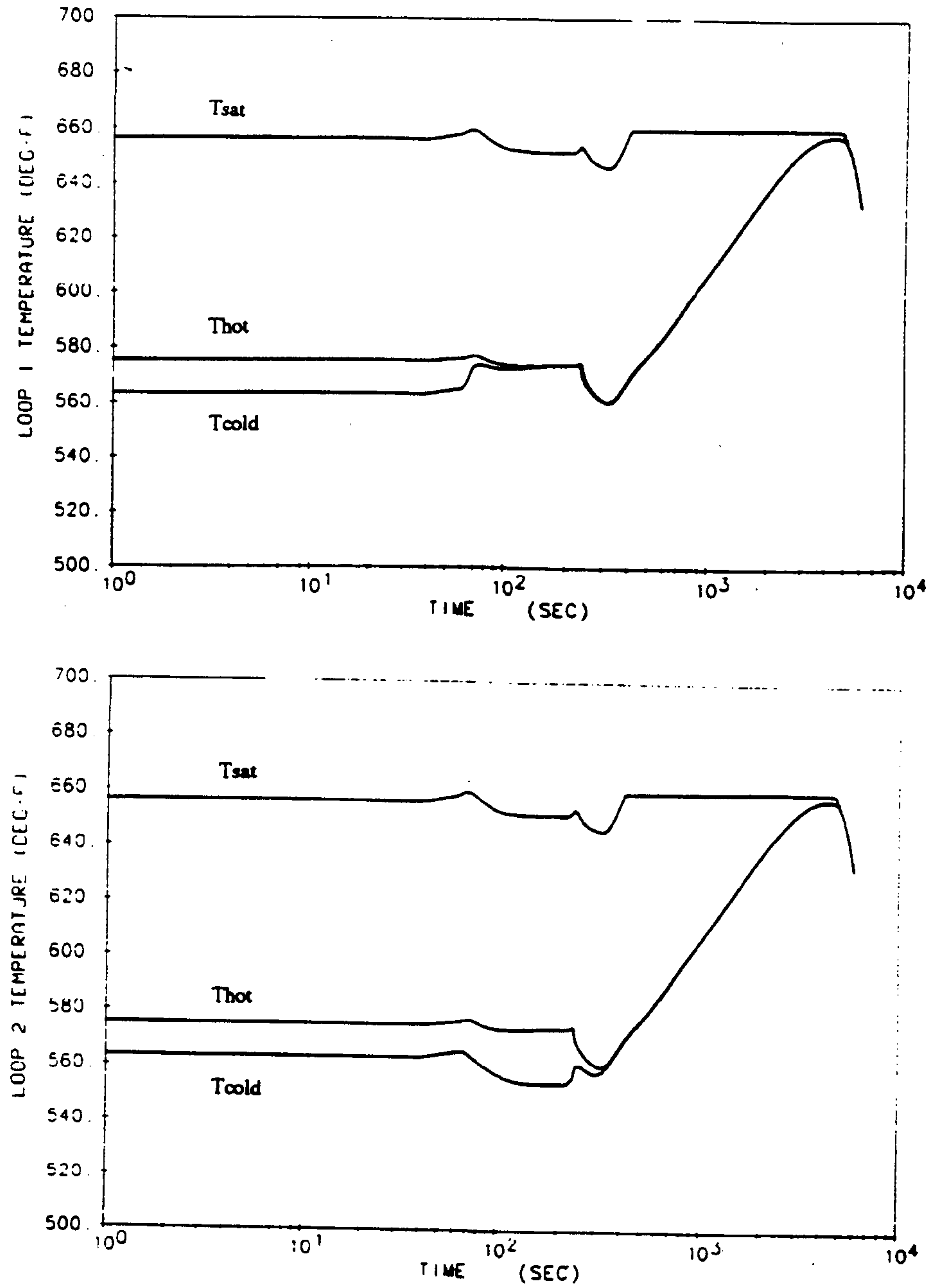


FIGURE 2.2.2.2-2

MAJOR RUPTURE OF A MAIN FEEDWATER PIPE AT 10 PERCENT POWER, WITH OFFSITE POWER - Hot Leg, Cold Leg, and Saturation Temperatures in the Faulted and Intact Loops

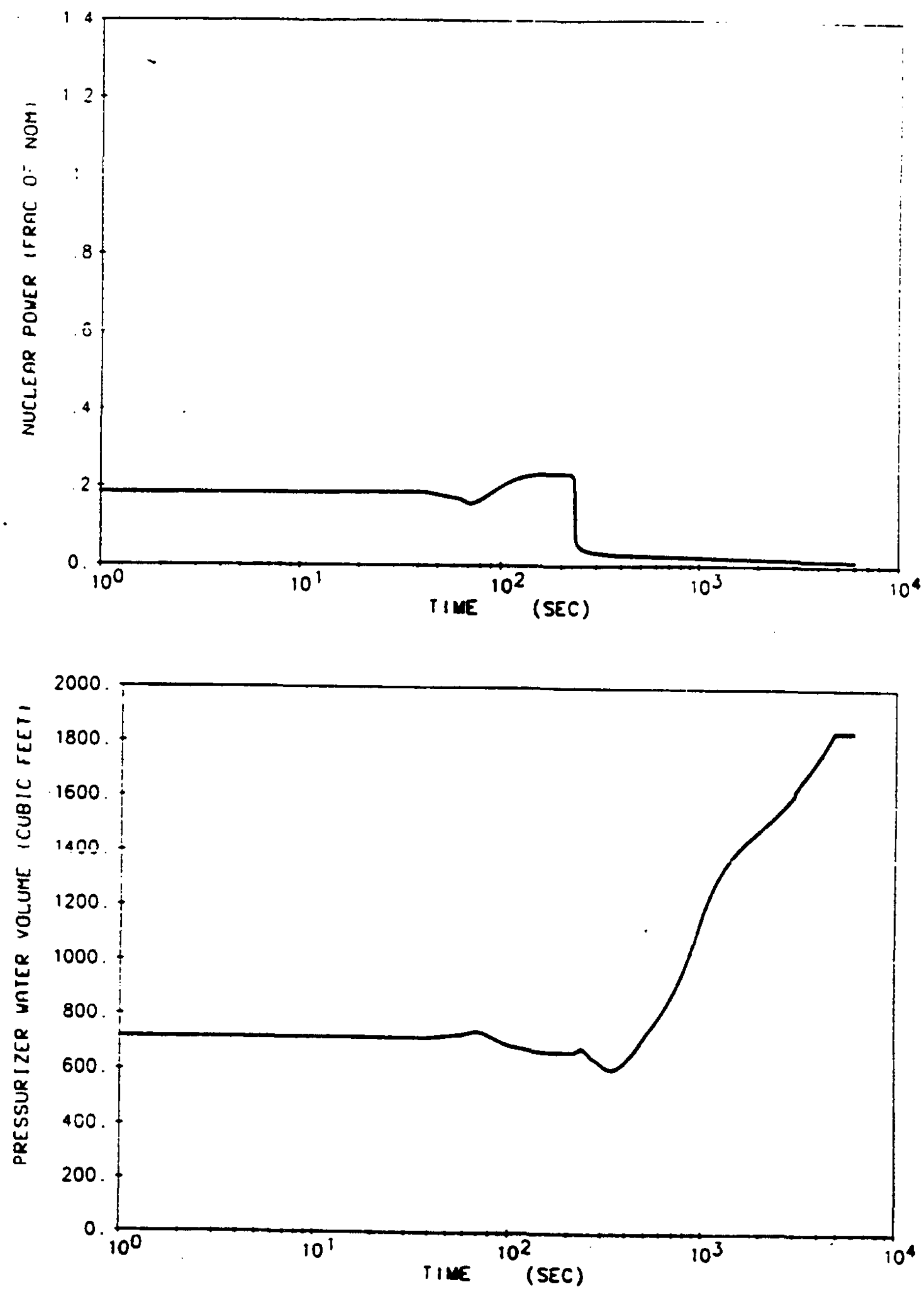


FIGURE 2.2.2.2-3

MAJOR RUPTURE OF A MAIN FEEDWATER PIPE AT 10 PERCENT POWER, WITHOUT OFFSITE POWER - Nuclear Power and Pressurizer Water Volume versus Time

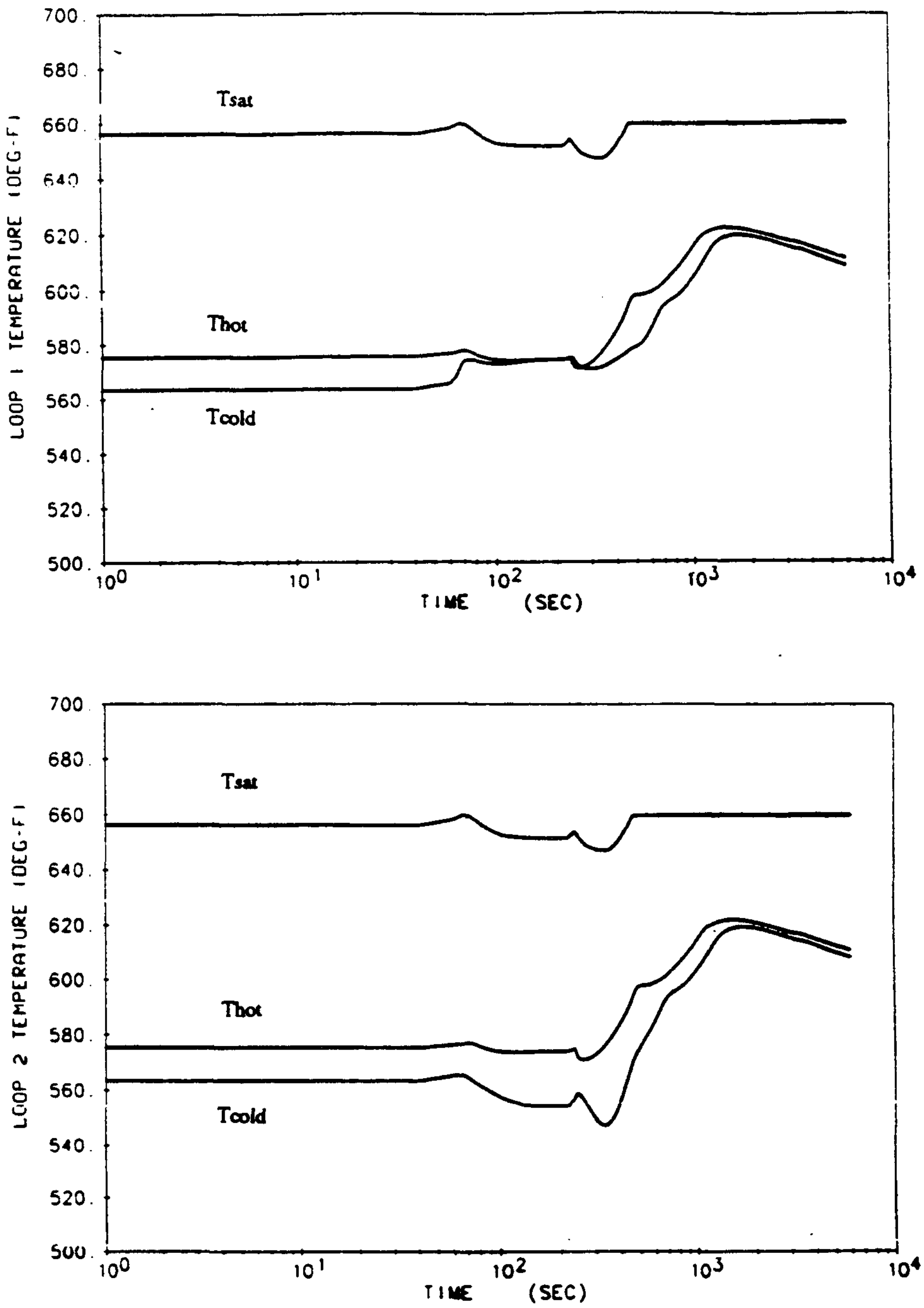


FIGURE 2.2.2.2-4

MAJOR RUPTURE OF A MAIN FEEDWATER PIPE AT 10 PERCENT POWER, WITHOUT OFFSITE POWER - Hot Leg, Cold Leg, and Saturation Temperatures versus Time

3.0 INSTRUMENTATION AND CONTROL SYSTEM DESIGN

3.1 Eagle 21 Process Protection System

3.1.1 Description

Process Instrumentation is comprised of those devices (and their interconnection into systems) which measure and process signals for temperature, pressure, fluid flow and fluid levels. Process instrumentation specifically excludes nuclear and radiation measurements. Process Instrumentation includes equipment which perform functions such as: process measurement, signal conditioning, dynamic compensation, calculations, setpoint comparison, alarm actuation, indication and recording. These functions are all necessary for day-to-day operation of the Nuclear Steam Supply System as well as for monitoring the plant and providing initiation of protective functions upon approach to unsafe plant conditions.

The Westinghouse Eagle 21 microprocessor based process protection upgrade system is applicable for those instrument systems which are "safety-related" as defined by IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations". The Eagle 21 portion of process instrumentation includes all necessary devices with the exception of sensors (transmitters and RTDs), indicators and recorders. Location of the process protection racks with respect to their interfaces within a typical nuclear power plant is depicted in Figure 3.1-1.

The Westinghouse Eagle 21 microprocessor-based process protection system is a functional replacement for existing analog process protection equipment used to monitor process parameters at nuclear generating stations and initiate actuation of the reactor trip and engineering safeguards systems. Features of the Eagle 21 equipment include the following:

A. Automatic surveillance testing to significantly reduce the time required to perform periodic surveillance tests.

B. Self calibration to eliminate rack drift and time consuming calibration procedures.

C. Self diagnostics to reduce the time required for troubleshooting.

D. Significant expansion capability to allow for rack consolidation and easily accommodate functional upgrades and plant improvements.

E. Modular design to allow for a phased installation into existing process racks and use of existing field terminations.

In a typical Eagle 21 Process Protection Instrument Channel, field sensors are connected to cabinet mounted terminal blocks. The process electronics power the sensors and perform signal conditioning, calculation and isolation operations on the input signals. However, each element of the process is not an individual electronic module or printed circuit board assembly. A multiple channel Eagle 21 Analog Input (EAI) module is used to power the field sensor(s) and perform signal conditioning. All calculations for the process channel functions are performed by a centralized Loop Calculation Processor (LCP). Typical functions performed by the LCP are as follows: summation, lead/lag, multiplication, comparison, averaging and square root conversion. Trip logic is provided through multiple channel Eagle

21 Partial Trip (EPT) output modules. Multiple channel isolated analog outputs are provided by Eagle 21 Analog Output (EAO) modules. In addition, all Eagle 21 process protection channels are configured to perform automatic surveillance testing via a centralized Test Sequence Processor (TSP).

The protection channels processed by the Eagle 21 process protection system are as follows:

A. Narrow Range Average Temperature and Delta Temperature (Hot and Cold)
B. Pressurizer Pressure
C. Pressurizer Water Level
D. Steam Flow and Feedwater Flow
E. Reactor Coolant Flow
F. Turbine Impulse Chamber Pressure
G. Steamline Pressure
H. Containment Pressure
I. Wide Range Reactor Coolant Temperature (Hot and Cold)
J. Wide Range Reactor Coolant Pressure
K. Refueling Water Storage Tank Level
L. Containment Sump Liquid Level
M. Narrow Range Steam Generator Water Level
N. Wide Range Steam Generator Water Level
O. Boric Acid Tank Level
P. Containment Spray Pump Discharge Header Flow
Q. Pressurizer Liquid Temperature
R. Pressurizer Vapor Temperature
S. Residual Heat Removal Pump Discharge Temperature

The Eagle 21 equipment has been designed to fit into the existing process racks and to preserve the interface with other plant systems, identical to the existing analog equipment configuration (Figure 3.1-2). The design maintains the existing field terminals to avoid new cable pulls or splices within the rack. The Eagle 21 equipment design also allows for rack consolidation of the process protection channels resulting in a reduction of the total number of racks required for a protection system upgrade. The components for each rack are built into subassemblies which are easily installed into the existing racks. All internal rack cabling is pre-fabricated. The subassemblies are tested in a factory mock-up to verify proper fit and operation. Detailed installation procedures and drawings are provided with each system.

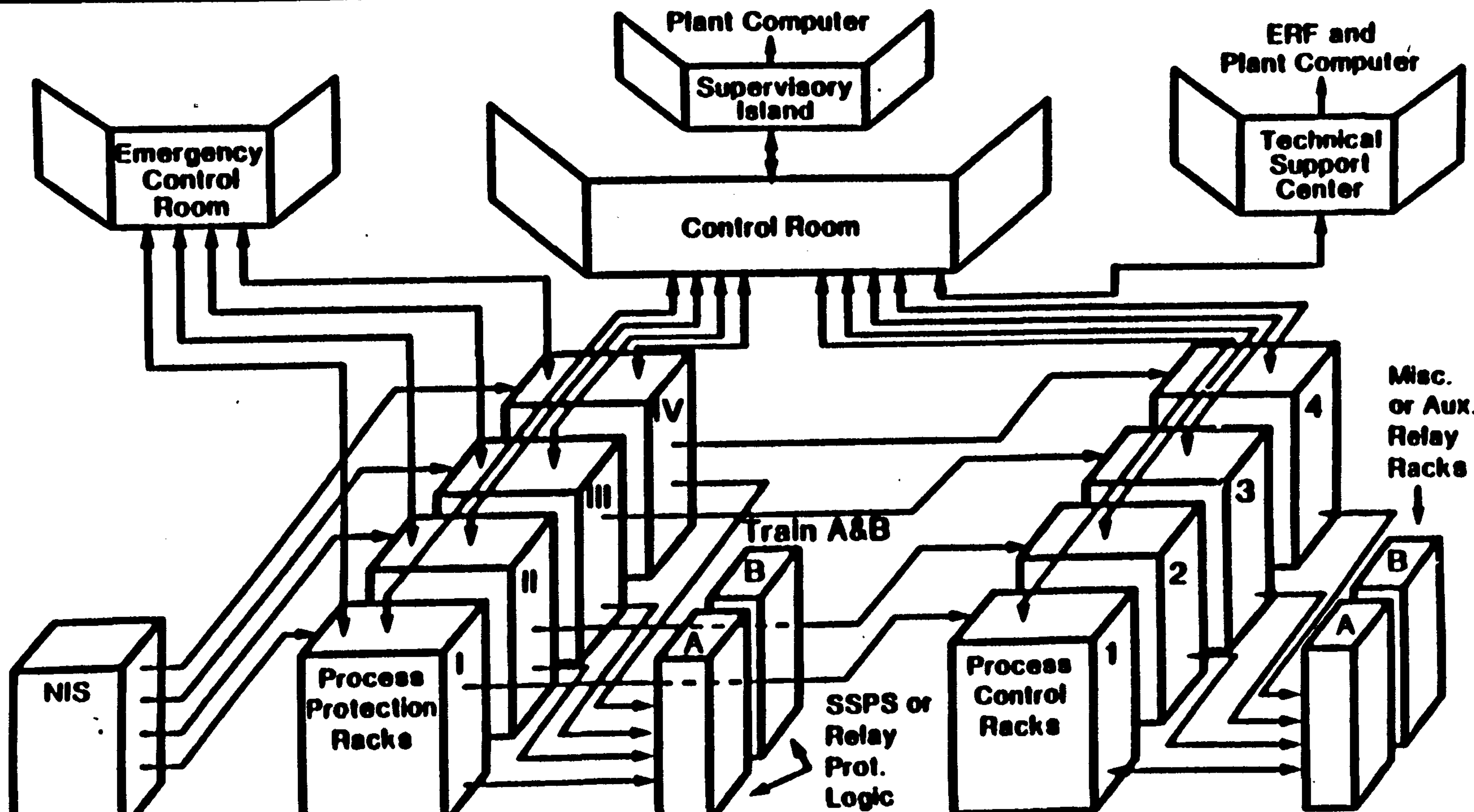
For a typical overview of the system design and features, refer to Reference 2.

3.2 <u>New Steamline Break Protection</u>

3.2.1 Description

The NEW STEAMLINE BREAK (NSLB) PROTECTION functional upgrade involves the replacement of the present steamline break protection with a new system. This new system reduces the potential for spurious actuation of Safety Injection (SI) at low power.

EAGLE-21 IMPLEMENTATION



Typical I&C System Block Diagram

FIGURE 3.1-1

EAGLE-21
DESIGN PHILOSOPHY

- Form, fit and function replacement

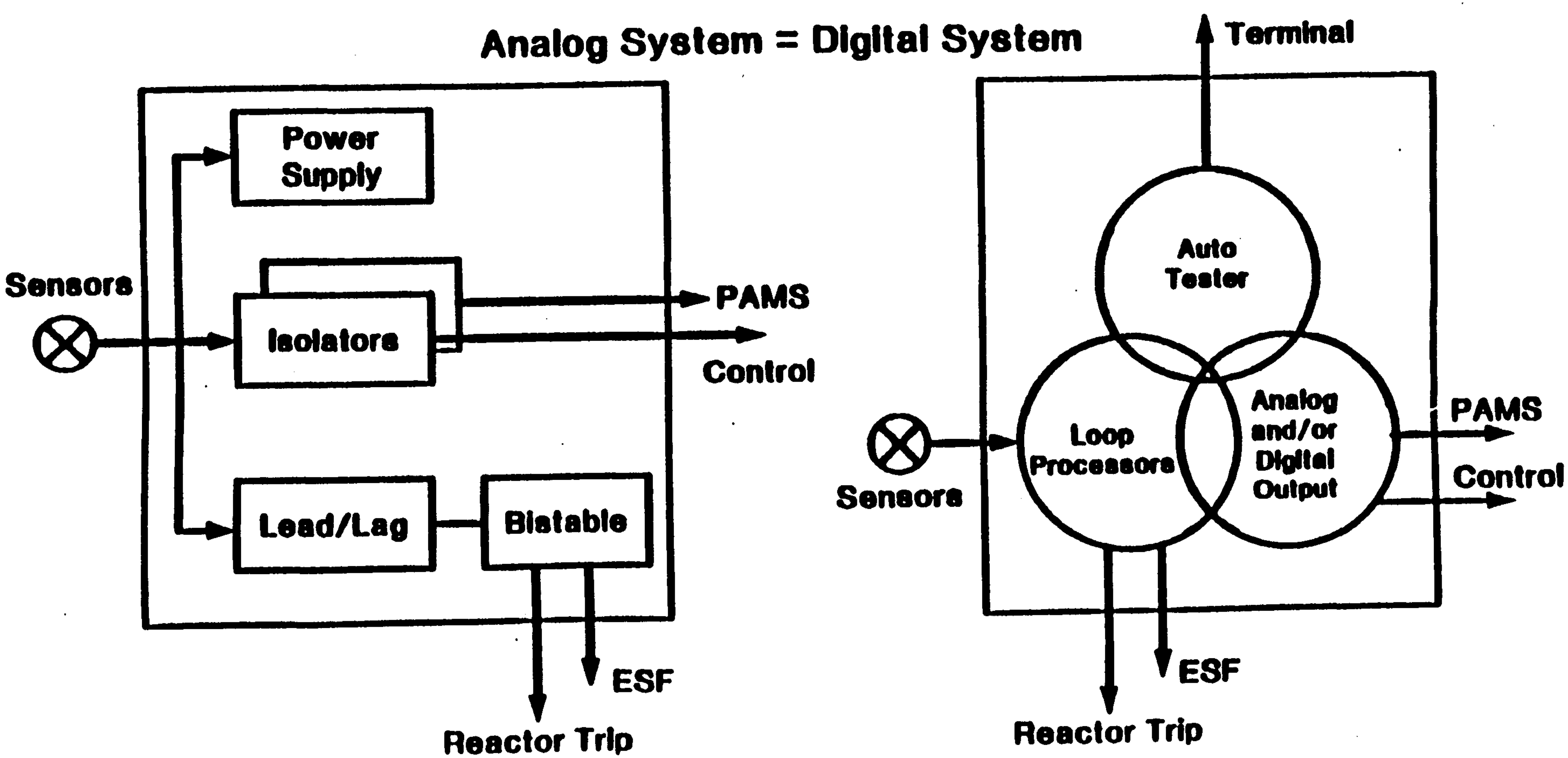


FIGURE 3.1-2

The old steamline break protection utilizes steamline flow, Lo-Lo TAVG and steamline pressure as process inputs (Figure 3.2-1, Sheet 1). The process protection system uses these signals to generate comparator (bistable) outputs for High Steamline Differential Pressure ($Px < Py - a$) and High Steamline Flow ($Fs > Fref$). The reactor protection system logic generates SI on High Steamline Differential Pressure (2 out of 3 coincidence logic) or High Containment Pressure (2 out of 3 coincident logic) and SI plus Steamline Isolation on High Steamline Flow (2 out of 4 coincident logic) coincident with Low Steamline Pressure (2 out of 4 coincidence logic) or Permissive P-12 Lo-Lo Tavg.

The NSLB PROTECTION (Figure 3.2-1, Sheet 2) modifies both the Reactor Protection Process and Logic systems. This upgrade requires the deletion, retention and addition of several functions. These changes are as follows:

Functions Deleted:
- SI and Steamline Isolation on high steamline flow coincident with P-12 Lo-Lo Tavg
- SI and Steamline Isolation on high steamline flow coincident with low steamline pressure
- SI on high steamline differential pressure

Functions Retained:
- Low pressurizer pressure SI
- All high containment pressure signals

Functions Added:
- SI and Steamline Isolation on 2 out of 3 coincidence of low steamline pressure
- Steamline Isolation on 2 out of 3 coincidence of high negative steamline pressure rate coincident with P-11 Pressurizer Pressure

3.2.2 Design and Implementation

Incorporating the NSLB protection results in the deletion of three functions (SI on high steamline differential pressure and Steamline Isolation plus SI on high steamline flow coincident with low steamline pressure or Permissive P-12 Lo-Lo Tavg) and the addition of two functions (Steamline Isolation on high negative steamline pressure rate and SI plus Steamline Isolation on low steamline pressure).

For the NSLB protection functional upgrade, the reactor protection system logic is modified to delete the SI on High Steamline Differential Pressure (Figure 3.2-2, Sheet 1) and Steamline Isolation plus SI on steamline flow coincident with low steamline pressure or P-12 Lo-Lo Tavg functions (Figure 3.3-2, Sheet 2). The NSLB protection logic requires the addition of a Steamline Isolation signal on 2 out of 3 coincidence of High Negative Steam Pressure Rate (rate-lag compensated) coincident with P-11 Pressurizer Pressure functions (Figure 3.2-3, Sheet 1) and SI and Steamline Isolation on 2 out of 3 coincidence of low steamline pressure (lead-lag compensated) (Figure 3.2-3, Sheet 2). With the addition of the High Steamline Pressure Rate signal, protection is provided to the plant when the plant is between cold and hot shutdown conditions. Figure 3.2-1, Sheets 1 and 2, shows the major additions and deletions to the Steamline Break Protection System.

In the reactor protection logic cabinets the existing 2 out of 3 high steamline differential pressure logic for actuation of SI is replaced with 2 out of 3 low steamline pressure logic (lead/lag compensated) to initiate SI and Steamline Isolation. Twelve (12) inputs from the process protection system comparators are used for either of the two logic arrangements, resulting in no net increase or decrease in input or testing relays. The logic arrangements use identical comparator signals and trip status lamps. A decrease of three (3) reactor trip first-out annunciators results from the implementation of the low steamline pressure logic.

Existing 2 out of 4 logic for the high steamline flow coincident with 2 out of 4 logic low steamline pressure or P-12 lo-lo Tavg for Steamline Isolation and SI is replaced with 2 out of 3 high steamline pressure rate logic (rate/lag compensated) to initiate Steamline Isolation. Each logic arrangement uses twelve (12) inputs from the process protection system comparators, resulting in no net increase or decrease in input or testing relays. The same number of computer signals and trip status lamps are used for either logic arrangement. One existing reactor trip first-out annunciator is replaced with a standard annunciator.

The Train A and B manual control board switches that are used to block Steamline Isolation and SI from high steamline flow coincident with low steamline pressure or low-low Tavg (P-12) are changed with the implementation of the NSLB protection. Included in the new protection logic is an interlock on pressurizer pressure low (P-11) to allow the operator to switch from the low steamline pressure protection to steamline pressure rate during normal heatup and cooldown operations. During this time, a manual block of Safety Injection and Steamline Isolation on Low Steamline Pressure is provided. Additionally, Steamline Isolation on High Negative Steamline Pressure Rate is permitted when the manual block has been initiated. When pressurizer pressure increases above the P-11 setpoint, the normal protection system is automatically reinstated. Above the P-11 Setpoint, Safety Injection and Steamline Isolation on Low Steamline Pressure are enabled and Steamline Isolation on High Negative Steamline Pressure Rate is defeated. Implementation of these changes does not require the associated computer signals and block status lamps to be modified.

The Eagle 21 system has been designed to perform automatic surveillance tests on the protection channels associated with the NSLB Protection functional upgrade.

3.2.3 Alarms, Annunciators and Status Lights

Additional control room alarms, annunciators and status lights are provided as part of the NSLB functional upgrade. These additional indications are as follows:

1. A trip status light is provided for each High Steam Pressure Rate Comparator (3 per loop) and for each Low Steamline Pressure Comparator (3 per loop). This status light informs the operator that a High Steam Pressure Rate or Low Steamline Pressure comparator has tripped for the channel in question.

2. An alarm and annunciator for each loop is actuated when any one of the High Steam Pressure Rate or Low Steamline Pressure channels in the loop trips (1 out of 3 logic). In addition, separate alarms and annunciators are provided (2 out of 3 coincident logic in any loop) to indicate that,
   a) initiation of Steamline Isolation on High Steam Pressure Rate below the P-11 Setpoint, and,
   b) initiation of Safety Injection and Steamline Isolation on Low Steamline Pressure above P-11.

EXISTING STEAMLINE BREAK PROTECTION SYSTEM

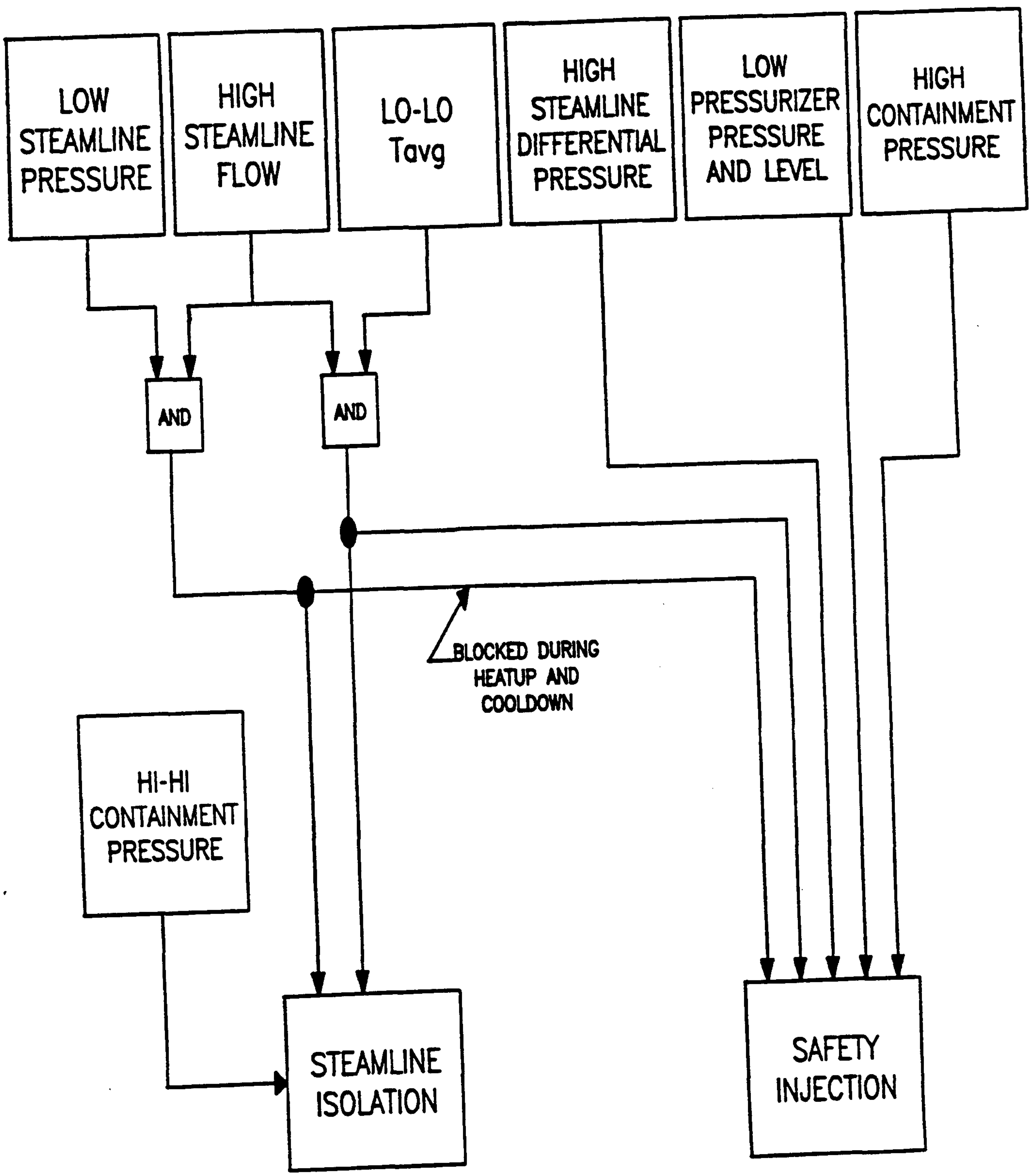


FIGURE 3.2-1
SHEET 1 OF 2

NEW STEAMLINE BREAK PROTECTION SYSTEM

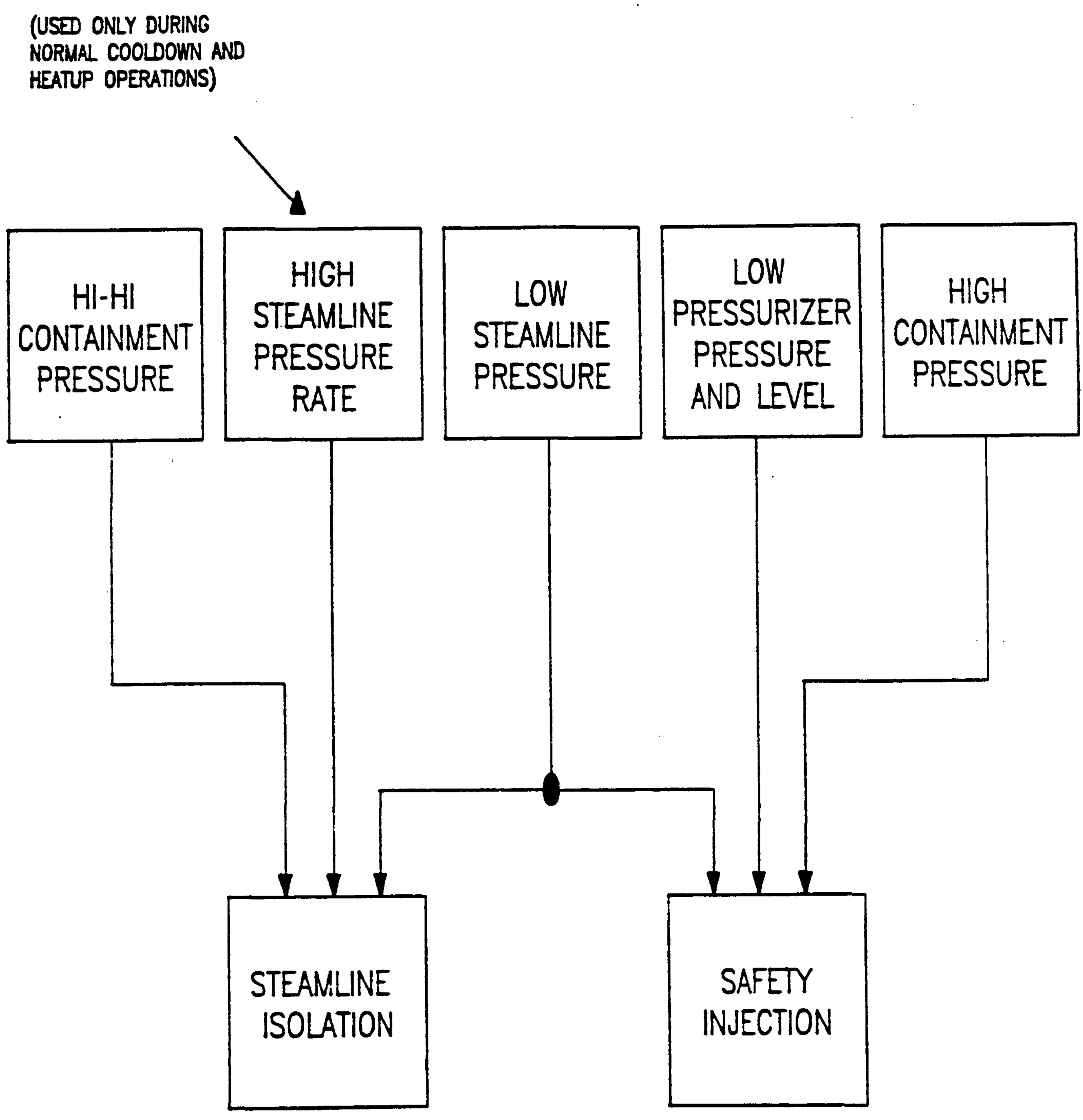


FIGURE 3.2-1
SHEET 2 OF 2

OLD STEAMLINE BREAK PROTECTION LOGIC (DELETED)

STEAMLINE DIFFERENTIAL PRESSURE

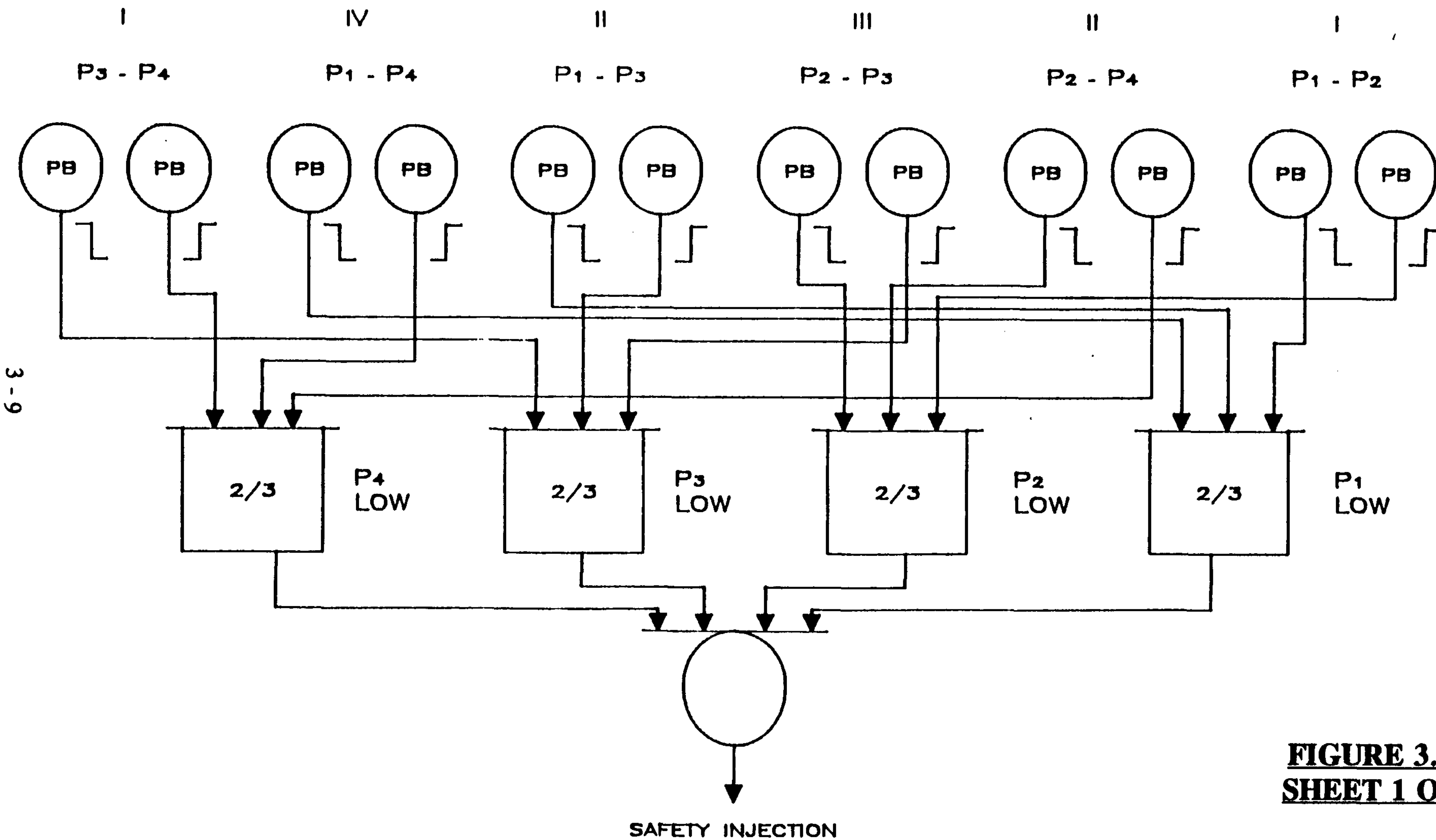


FIGURE 3.2-2
SHEET 1 OF 2

OLD STEAMLINE BREA ROTECTION LOGIC
(DELETED)

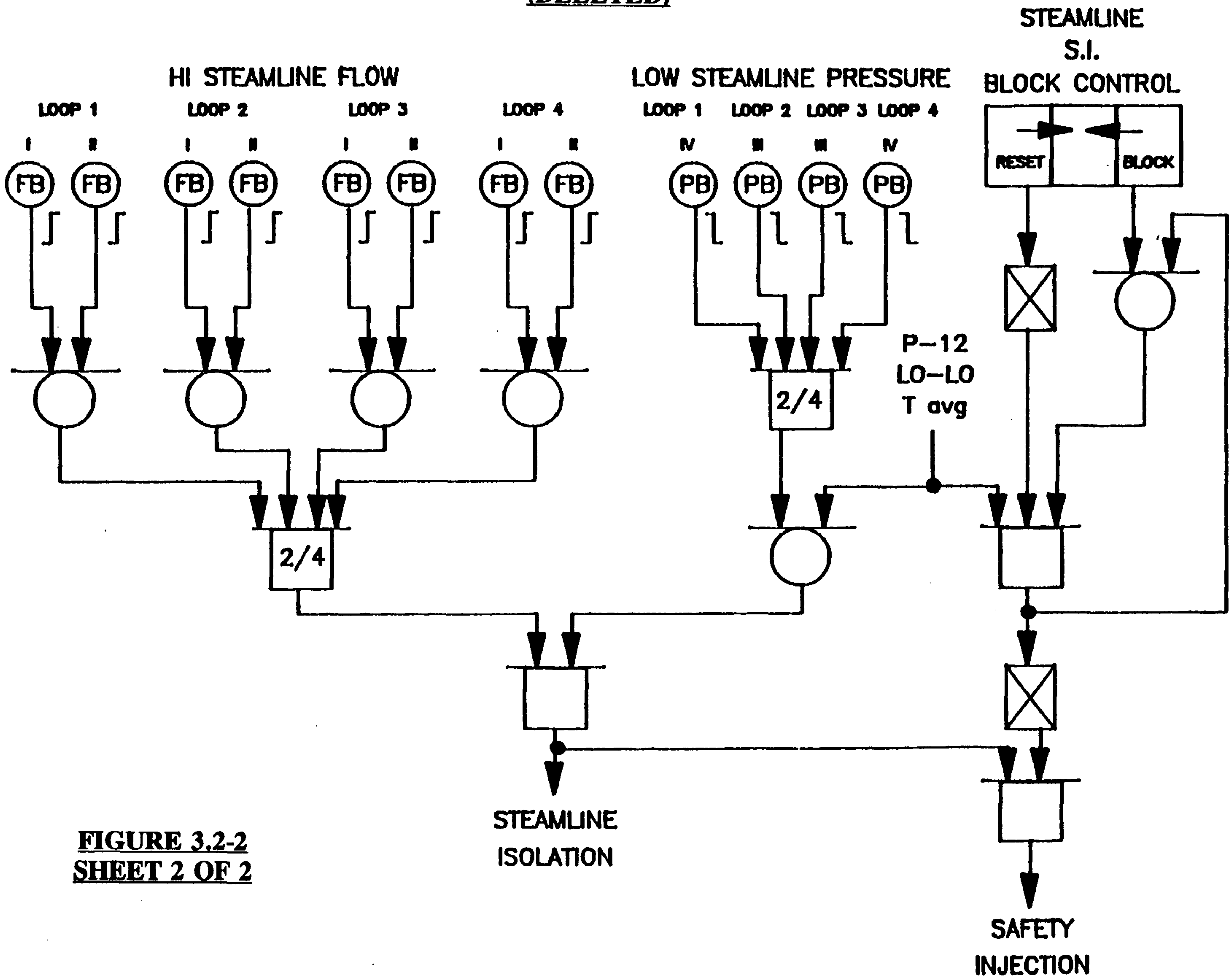


FIGURE 3.2-2
SHEET 2 OF 2

NEW STEAMLINE BREAK PROTECTION LOGIC
HIGH STEAM PRESSURE RATE (RATE - LAG COMPENSATED)

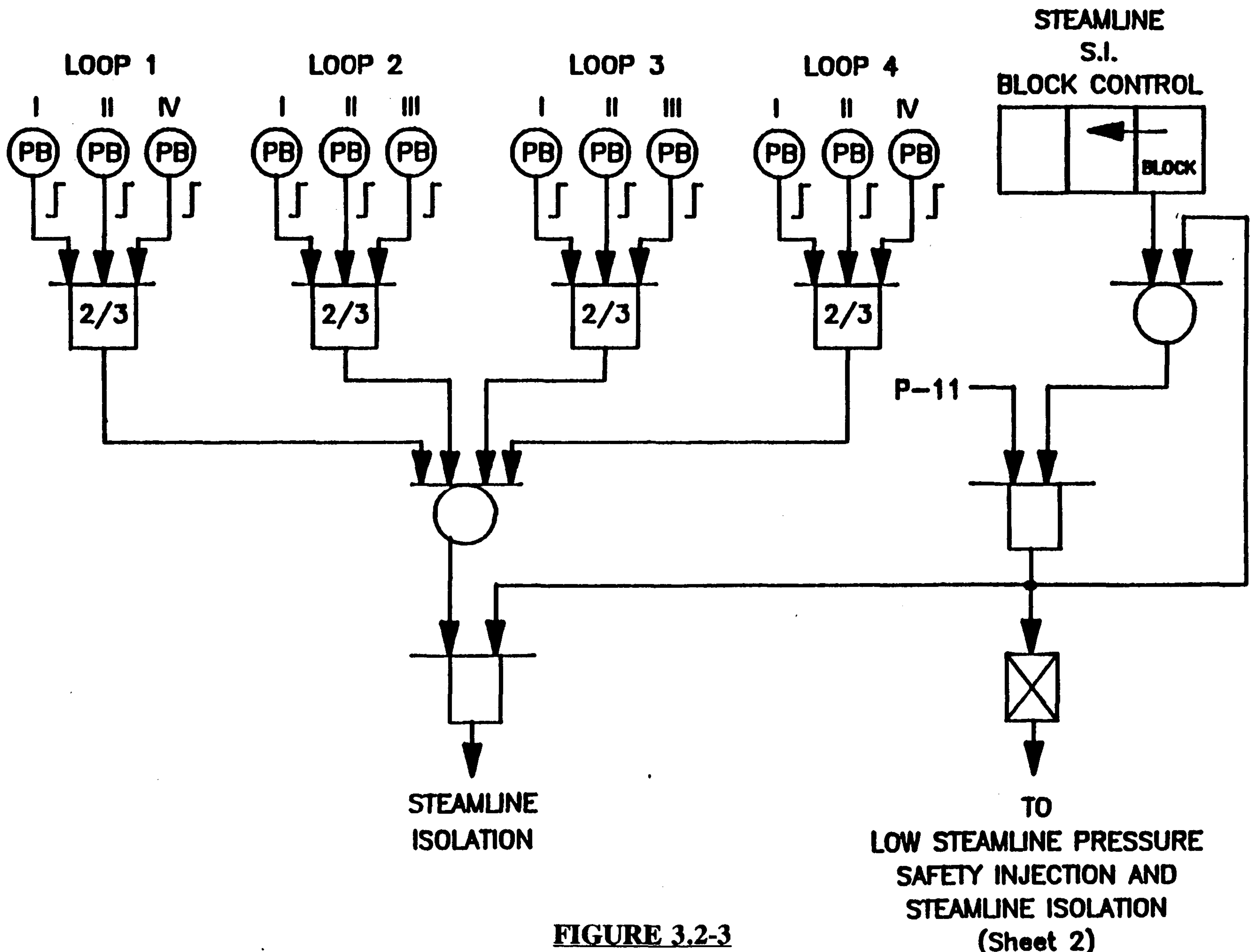


FIGURE 3.2-3
SHEET 1 OF 2

NEW STEAMLINE BREAK PROTECTION LOGIC
LOW STEAMLINE PRESSURE (LEAD-LAG COMPENSATED)

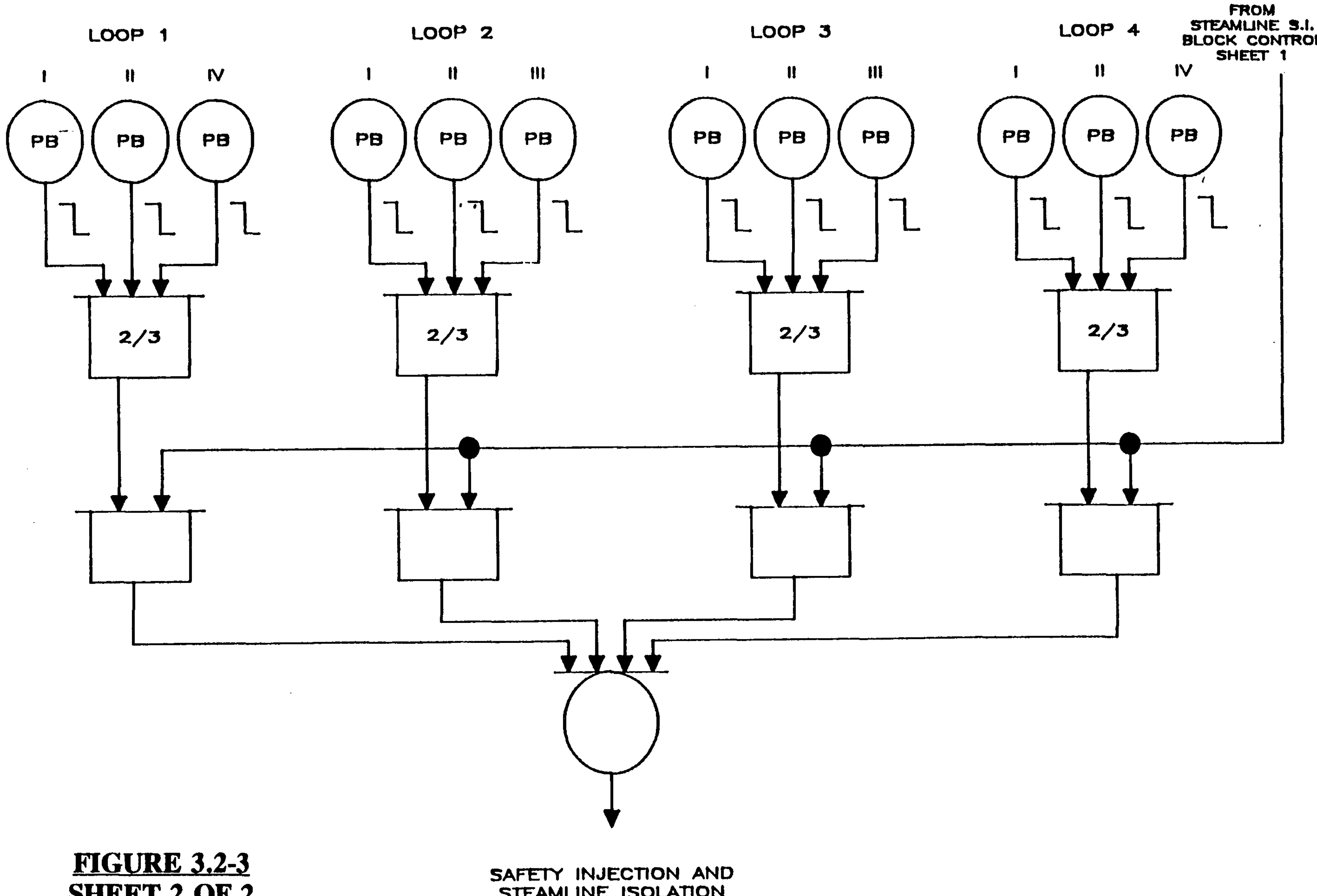


FIGURE 3.2-3
SHEET 2 OF 2

3.3 Trip Time Delay (TTD)

3.3.1 Description

In June 1985, the Westinghouse Owners Group (WOG) established a Trip Reduction and Assessment Program (TRAP) to address the announced goal of the Nuclear Utility Management and Resources Committee (an industry oversight group) to achieve a "significant" reduction in unnecessary reactor trips. The WOG TRAP commissioned a review of operating experience between 1979 and 1985 at Westinghouse supplied plants, which revealed that the largest number of automatic reactor trips (about 40 percent) were associated with the feedwater system.

This review was followed by several feasibility studies to identify and estimate possible sources of available margin, which could be applied to reduce the frequency of unnecessary feedwater system related reactor trips. Several approaches were identified, one of which was Trip Time Delay (TTD). This approach delays the steam generator low-low level protection functions (i.e., reactor trip and auxiliary feedwater system actuation) at low power levels or start-up operations.

3.3.2 Design and Implementation

The TTD acts to delay the reactor trip and auxiliary feedwater system actuation when the low-low steam generator water level trip setpoint is reached. The reason for the delay is to allow time for corrective action by the operator or for natural stabilization of shrink/swell water level transients. The TTD delays the steam generator low-low level reactor trip signal during low power or start-up operations. The trip delays are based upon the prevailing power level at the time that the low-low water level trip setpoint is reached in any of the steam generators and the number of faulted steam generators (Figure 3.3-1). These delay times are longer at lower power versus high power. The delays are also longer when only one steam generator has a low-low level condition as opposed to more than one steam generator.

When the low-low level trip setpoint is reached in any of the steam generators, the low-low level trip signal is not immediately transmitted to the Solid State Protection System (SSPS). Instead, it is delayed for a period of time, which is determined from a specified trip delay versus power level relationship. Power level is derived from the temperature difference between the hot and cold legs of the reactor coolant system (DELTA-T) at the time the low-low water level trip setpoint is reached. If the power level drops during the trip delay interval, the TTD logic does not lengthen the trip delay. However, if power level rises, the trip delay is shortened to correspond with the increased power level.

If the water levels in all the steam generators are not restored to levels above the low-low level trip setpoint before the trip delay is elapsed, then the TTD logic transmits the low-low level trip signals into the SSPS channel logic, where the reactor trip and auxiliary feedwater system initiation are actuated.

The Eagle 21 system has been designed to allow the Narrow Range Thot and Tcold RTDs for Thermal Overpower and Overtemperature Delta-T protection to be tested without disturbing the TTD channel. Thus, there is no need for the associated Steam Generator Low-Low Level channels to default to a zero time delay setpoint when the Thermal Overpower and Overtemperature Delta-T channels are in test.

The conceptual design of the TTD is described in Reference 1. This report was reviewed by the NRC staff and approved for use. In January 1988, the staff issued a Safety Evaluation Report (SER) approving the conceptual design of TTD. The NRC staff found the report acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the report and the SER.

The information contained in the 1988 SER will be used as a basis for licensing TTD. As noted in the WCAP, the conceptual design of the TTD consisted of the following:

1. All logic decisions were implemented in the SSPS.

2. Each protection set was represented by four (4) steam generator channels.

3. Nuclear Flux was used as an input to TTD to enable the trip time delay select logic.

4. A series of discrete timers were installed to provide the time delays when setpoints were reached.

Several differences exist between the conceptual design and the Eagle 21 functional upgrade of TTD. These include the following:

1. The design and all logic decisions will be implemented in the Eagle 21 process protection system.

2. Where Nuclear Flux was used as an input to the TTD logic, this input is replaced by Delta-T due to the increased uncertainties associated with the nuclear flux detectors at low power levels.

3. The discrete analog timers are replaced with microprocessor controlled continuous time delays. By using these programmed time delays rather than the discrete timers, the signal path is uninterrupted allowing for a more accurate signal to be processed.

Although these differences exist, the changes to the overall conceptual design of the TTD as discussed in the WCAP will not impact the conclusions reached in the SER due to the results derived from the re-analysis of the accident analysis as described in Section 2.3.

In support of the licensing of the TTD for Watts Bar Eagle 21, it should be noted that the TVA - Sequoyah TTD implementation was reviewed by the NRC staff is functionally equivalent to the Watts Bar implementation with the exception that in the Sequoyah application only 2 steam generator level channels represent Protection Sets I and II, while 4 steam generator level channels represent Protection Sets III and IV. During the design of the Watts Bar TTD, a conscious decision was made to attempt to make the Watts Bar implementation similar to the conceptual design evaluated. Therefore, several of the steam generator level channels were relocated to allow 4 steam generator level channels to represent three protection sets.

3.3.3 Alarms, Annunciators and Status Lights

Additional control room alarms, annunciators and status lights are provided as part of the Trip Time Delay functional upgrade. These additional indications are as follows:

1. A low-low level alarm and annunciator window will be provided for each Steam Generator to signify that the water level in at least one channel has dropped below the low-low level setpoint in that steam generator and that the TTD time delays have started. When this condition is indicated, the operator will then observe the individual steam generator level channel indicators to determine if the one-out-of-four or the two-out-of-four time delay is in effect.

2. No new status lights are provided as part of the TTD functional upgrade.

TRIP TIME DELAY (FOUR STEAM GENERATORS)

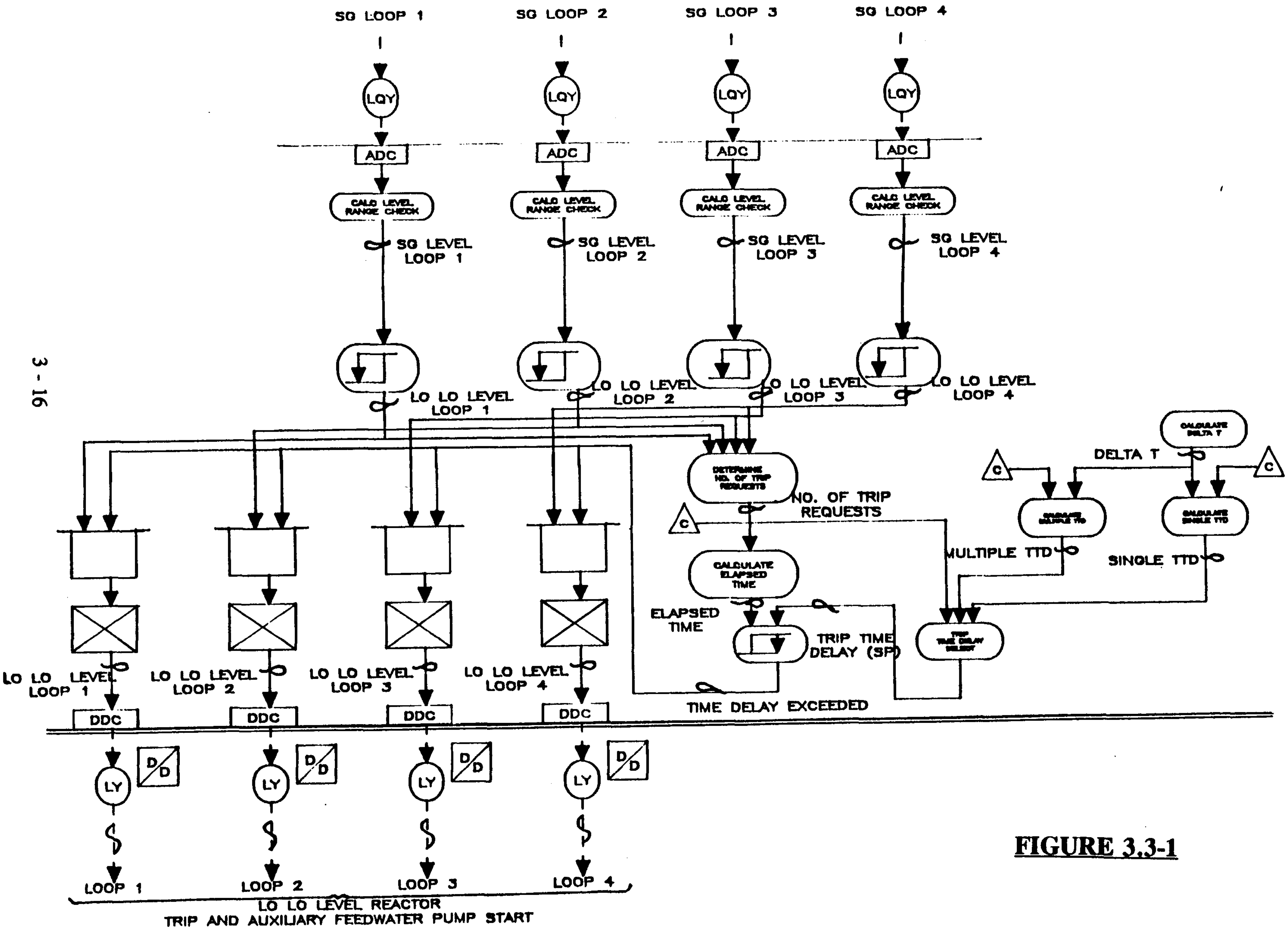


FIGURE 3.3-1

3.4 Elimination of Low Feedwater Reactor Trip Median Signal Selector

3.4.1 Description

The basic function of the reactor protection circuits associated with steam generator low-low water level trip channels is to preserve the steam generator as a heat sink for residual heat removal. This automatic protective action is taken before the steam generators are dry to maintain the heat sink, reduce the capacity and starting time requirements of the Auxiliary Feedwater System and to minimize the thermal transient on the Reactor Coolant System. This trip is actuated on coincidence of two out of three low-low water level signals in any steam generator. This modification is not part of the Eagle 21 process protection system, but will be used by the Watts Bar plant as part of the Westinghouse Owners Group (WOG) Trip Reduction Program to enhance the reliability of certain plant control systems by eliminating the Low Feedwater Flow Reactor Trip. Elimination of this trip, increases plant availability, reduces challenges to the protection system and provides resolution to control/protection interaction concerns.

Originally, the low feedwater flow reactor trip was used in conjunction with the steam generator low-low water level reactor trip as a means of satisfying IEEE Std. 279-1971, Control and Protection System Interaction (CPI) criteria (Figure 3.4-1). Paragraph 4.7.3 of the standard states: "Where a single random failure can cause a control system action that results in a generating station condition requiring protective action and can also prevent proper action of a protection system channel designed to protect against the condition, the remaining redundant protection channels shall be capable of providing the protective action even when degraded by a second random failure." At the Watts Bar plant, three channels per protection set are used for the low-low level reactor trip (e.g., L-517, L-518 and L-519). Two of these channels (L-517 and L-518) are used for the low-level/low feedwater flow diverse reactor trip leaving the remaining channel (L-519) as the only channel available for control.

In a situation where a control signal is derived from a protective function, the possibility arises that a single, failed protection system channel could be the initiating event of a transient requiring protective action. Thus to maintain immunity of the protection system to a single failure, it is necessary either to design a control system that is not affected by protection system failures or design the protection system such that it is immune to a second failure, assuming that one of these can influence the control system. These principles are the intent of Section 4.7.3 of IEEE Std. 279-1971.

The intent of Section 4.7.3 (CPI) is satisfied with the implementation of a Median Signal Selector (MSS). With the MSS installed, a failed instrument channel cannot cause a control system action which will initiate a plant transient requiring protective action. Since no adverse control system action may now result from a single, failed protection system instrument channel, a second random protection system failure (as would otherwise be required by IEEE Std. 279-1971) need not be considered. Thus the diverse low level/low feedwater flow trip is no longer required for a defense against control/protection interaction.

3.4.2 Design and Implementation

To eliminate the low feedwater flow reactor trip and improve the reliability of the Feedwater Control System, a MSS for each steam generator is installed in the plant process control system (Figure 3.4-2). The MSS selects the median of three steam generator narrow range level input signals which prevents a single failed protection system instrument channel from causing a control system transient requiring protective action. Additionally, challenges to the reactor protection system are reduced which results in plant availability being increased due to the minimization of fatigue buildup on critical components. In this role, the MSS serves as a functional isolator between the Control and Protection systems thereby eliminating the need for the low feedwater flow reactor trip which existed to satisfy the CPI criteria. The types of failures and human errors that typically initiate control system upsets in existing system designs which may be prevented by using the MSS are:

1. Failure of a sensor that is aligned to provide an input signal to the control system.

2. Malfunction of a process protection channel that provides an input signal to the control system.

3. Operator/technician error when performing surveillance test on a process channel aligned to the control system without switching to manual control.

The MSS devices are physically located within the Process Control System racks. The reasons for this location are three-fold. First, no single protection channel set has enough information to determine if a single electrical signal for a given process variable is valid. To provide the necessary information would require a considerable increase in the amount of inter-channel communication among the protection channel sets which would lead to concerns about the independence of the redundant portions of the protection system. Second, is that the MSS not relied upon to perform a safety function such as reactor trip or engineered safety features actuation. Third, a failure of the MSS does not compromise the ability of the protection system to perform its safety related functions (i.e., failure of the MSS will not disable a protection system channel).

For a detailed description of the design, implementation and conformance to the applicable licensing criteria for the MSS and the elimination of the Low Feedwater Flow Reactor Trip, refer to Reference 3.

3.4.3 Alarms, Annunciators and Status Lights

All alarms, annunciators and trip status lights have been removed with the elimination of the Low Feedwater Flow Reactor Trip. No new indications provided to the operator as a result of the installation of the MSS.

3.5 References

1. Miranda, S., LaMuro, R. C., McHugh, C. J., "Steam Generator Low Water Level Protection System Modifications to Reduce Feedwater-Related Trips", WCAP-11325-P-A Rev. 1, February 1988.

2. Erin, L. E., "Topical Report Eagle 21 Microprocessor Based Protection System", WCAP-12374 Rev. 1, December 1991.

3. Mermigos, J. F., "Median Signal Selector for Foxboro Series Process Instrumentation, Application to Deletion of Low Feedwater Flow Reactor Trip", WCAP-12417, October 1989.

PRESENT FUNCTIONAL DESIGN

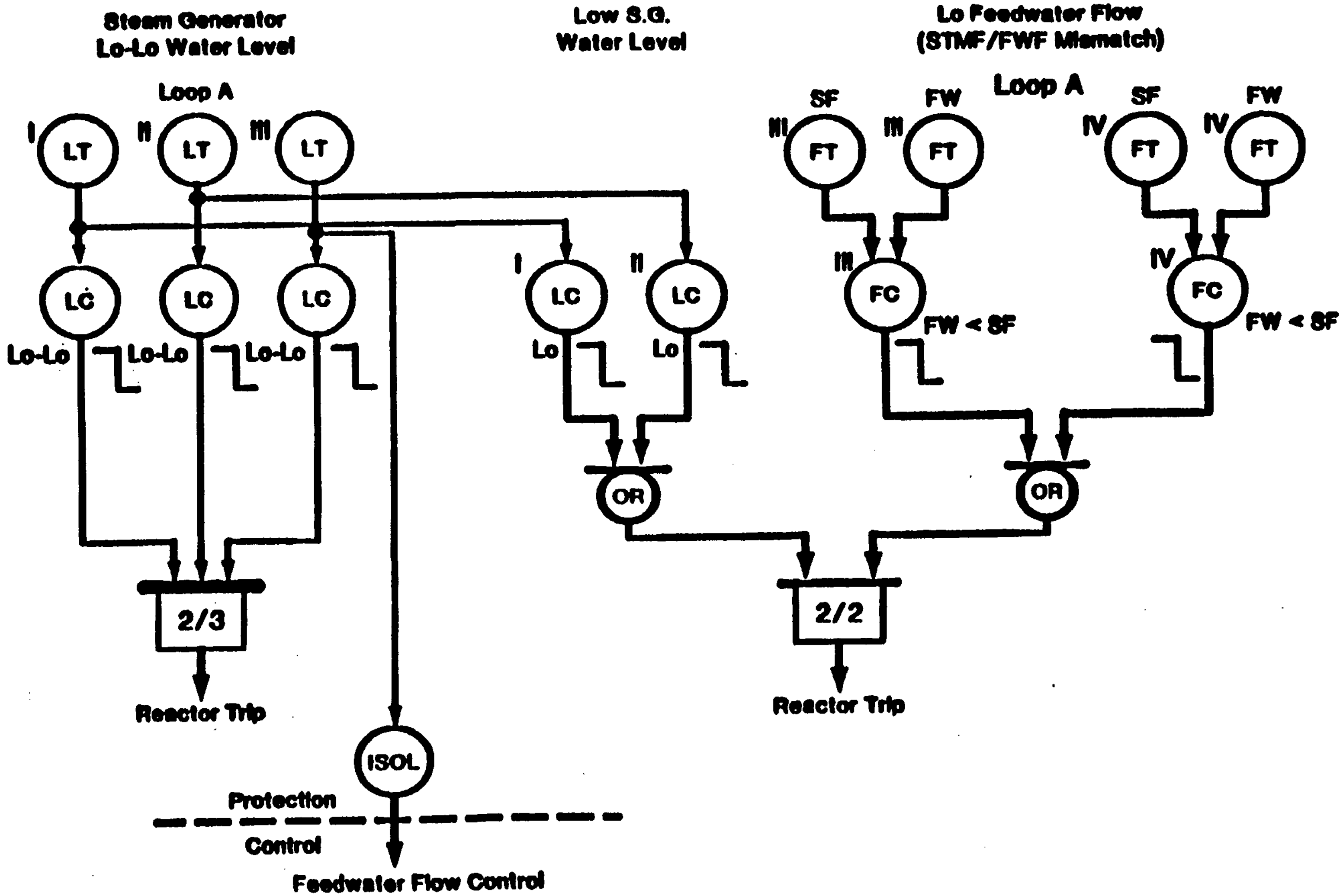


FIGURE 3.4-1

MEDIAN SIGNAL SELECTOR
FUNCTIONAL DESIGN

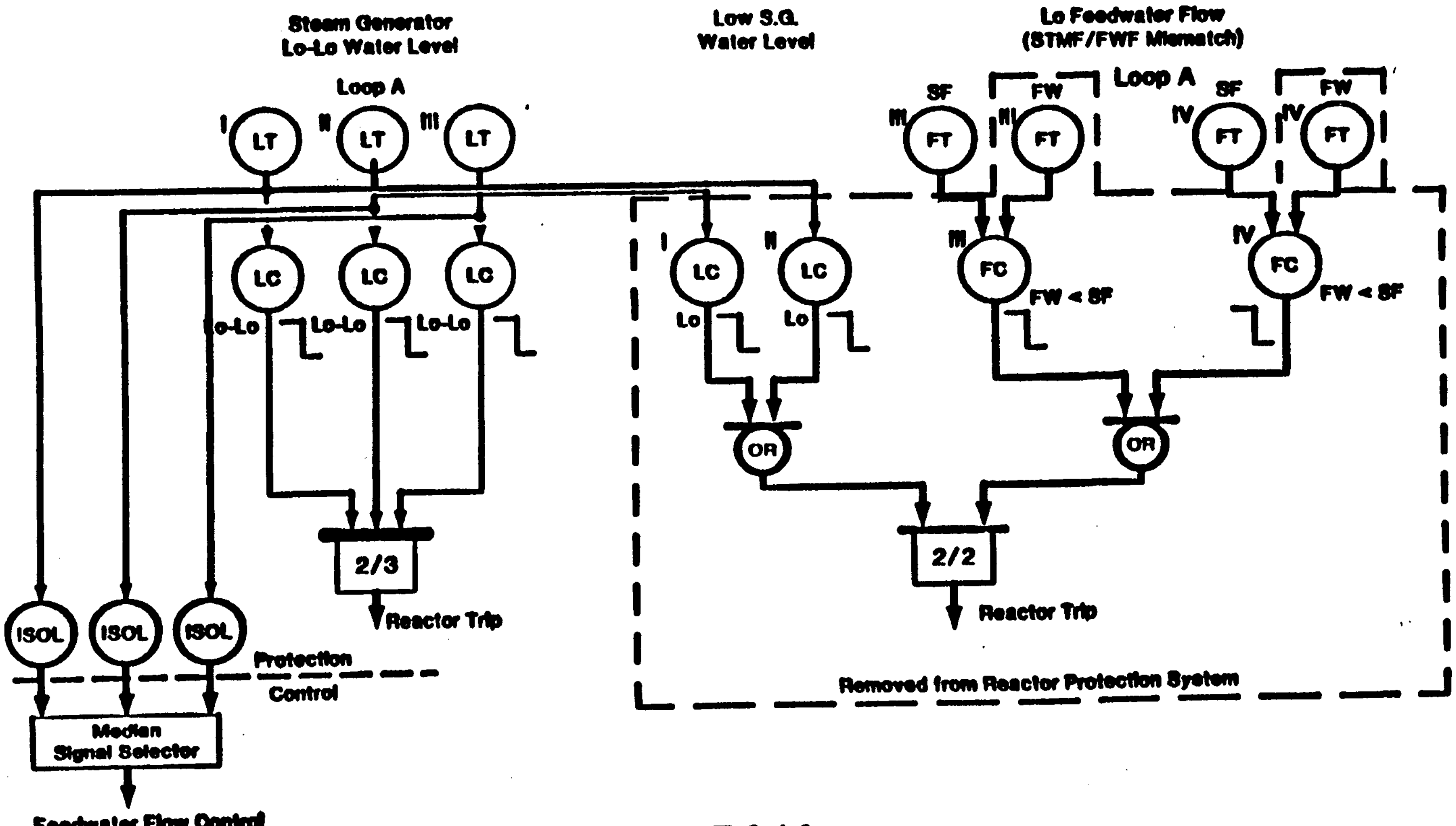


FIGURE 3.4-2

3.6 Control Systems

3.6.1 Control Systems Setpoint Evaluation

Replacement of the existing protection system with Eagle-21 hardware and elimination of the RTD bypass loops in the Watts Bar units, introduces additional signal processing delays that were reviewed to determine if a modification of control system setpoints was needed in order to prevent the challenging of safety system setpoints in the event of certain unit transients; e.g., large load rejections. It was determined that, with the addition of a lead-lag function on the measured Delta-T, the remaining gains and time constants for the Overpower and Overtemperature Delta-T equations have not changed from the current version of the Watts Bar Technical Specifications. Therefore, the Technical Specification values currently in-force, together with the values for the new lead/lag were used to evaluate the margin-to-trip.

The transient used to evaluate the margin-to-trip was a 50% load rejection from full power. For all of the cases analyzed (different initial rod positions, ramp and step load changes, etc.), a sufficient margin to an Overtemperature Delta-T was determined, which would preclude a challenge to the safety systems. The overtemperature trip was the most limiting of the Delta-T protection functions.

3.6.2 LTOPS Setpoint Evaluation

The wide range pressure signals used by the Low Temperature Overpressure Protection System (LTOPS) will be processed through the Eagle-21 system. The Eagle-21 will replace the old Foxboro protection gear, resulting in a nominal additional instrument time delay of 250 ms. The peak pressures following an overpressure event are sensitive to instrumentation time delays, so that the overpressures determined from previous analyses are no longer a viable basis for setpoint determination.

The setpoint program specified in Table 3.6-1, and shown in Figures 3.6.1 and 3.6.2 includes the impact of the added process instrument delay time resulting from Eagle-21 implementation, as well as the reactor vessel pressure-temperature limits based on Revision 2 of NRC Reg. Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials". The pressure-temperature limits are based on a reactor vessel exposure of 7 Effective Full Power Year (EFPY) for Watts Bar Unit 1 and 32 EFPY for Watts Bar Unit 2. The temperature dependent setpoint function also accounts for 50°F thermal transport effect (steam generator tube water swept pass the cold leg RTD), and a 27°F temperature streaming and instrumentation uncertainty. Consistent with current Westinghouse practice, the setpoint program assumes nominal values and steady-state pressure-temperature limits.

The setpoint have been selected to prevent opening both PORV's at the same time, while providing reactor coolant pump number 1 seal protection. Additionally, PORV setpoint pressure was selected to be above the residual heat removal system relief valve setpoint of 450 psig.

Table 3.6-1

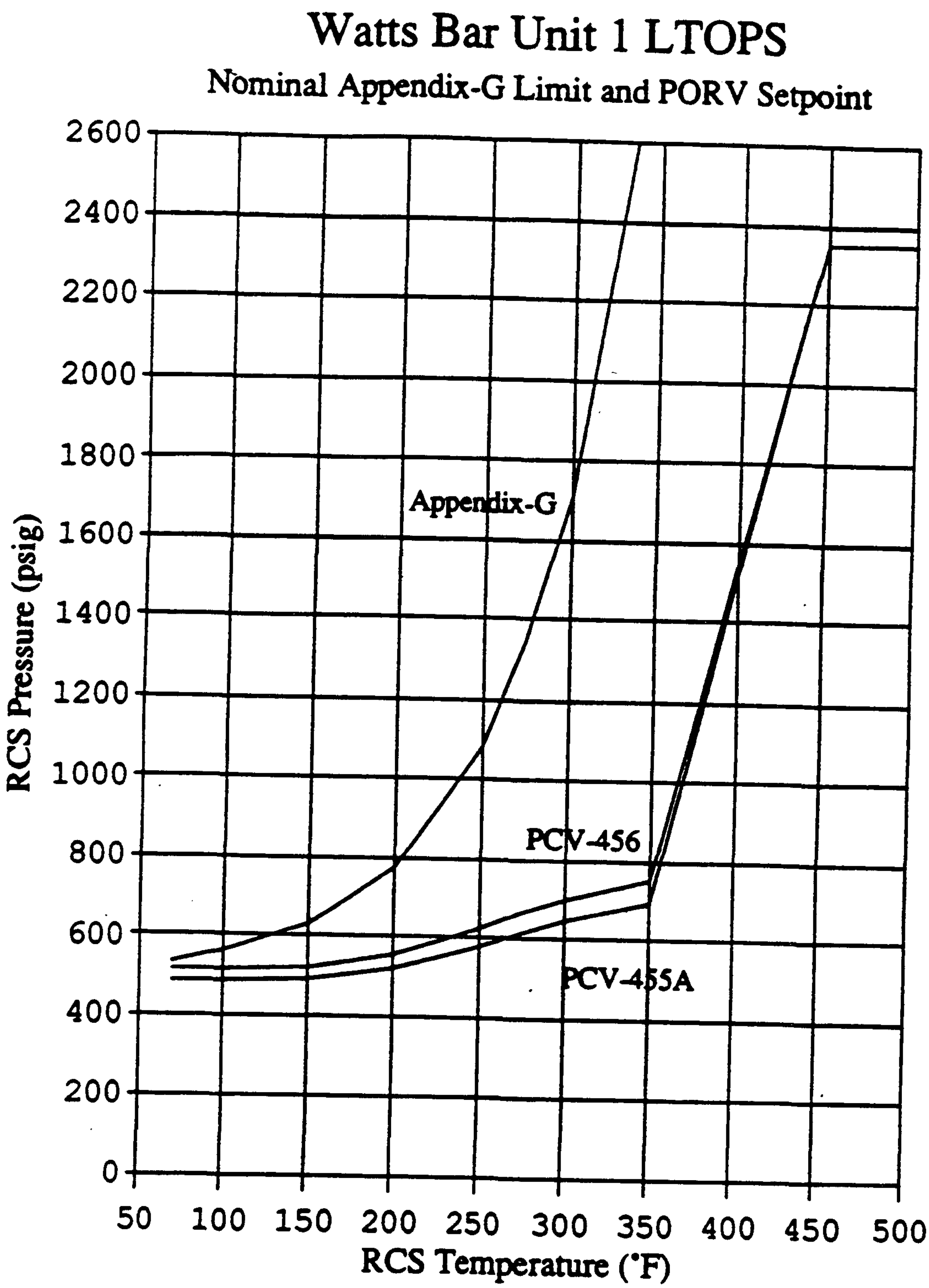
Table 1 Watts Bar Units LTOPS Setpoint Program

Watts Bar Unit 1 (WAT) LTOPS Setpoints

RCS Temp (°F)	PORV Setpoint (psig)	
	PCV-455A	PCV-456
70.0	485.0	515.0
100.0	485.0	515.0
150.0	490.0	520.0
200.0	520.0	555.0
250.0	580.0	625.0
275.0	615.0	665.0
300.0	652.0	702.0
350.0	693.0	748.0
450.0	2350.0	2350.0

Watts Bar Unit 2 (WBT) LTOPS Setpoints

RCS Temp (°F)	PORV Setpoint (psig)	
	PCV-455A	PCV-456
70.0	565.0	605.0
100.0	570.0	610.0
150.0	585.0	625.0
175.0	620.0	675.0
200.0	645.0	700.0
250.0	655.0	710.0
300.0	670.0	725.0
350.0	670.0	725.0
450.0	2350.0	2350.0



Watts Bar Unit 1 LTOPS
Nominal Appendix-G Limit and PORV Setpoint
RCS Pressure (psig)
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
2600
Appendix-G
PCV-456
PCV-455A
50
100
150
200
250
300
350
400
450
500
RCS Temperature (°F)

Figure 3.6-1

Watts Bar Unit 2 LTOPS

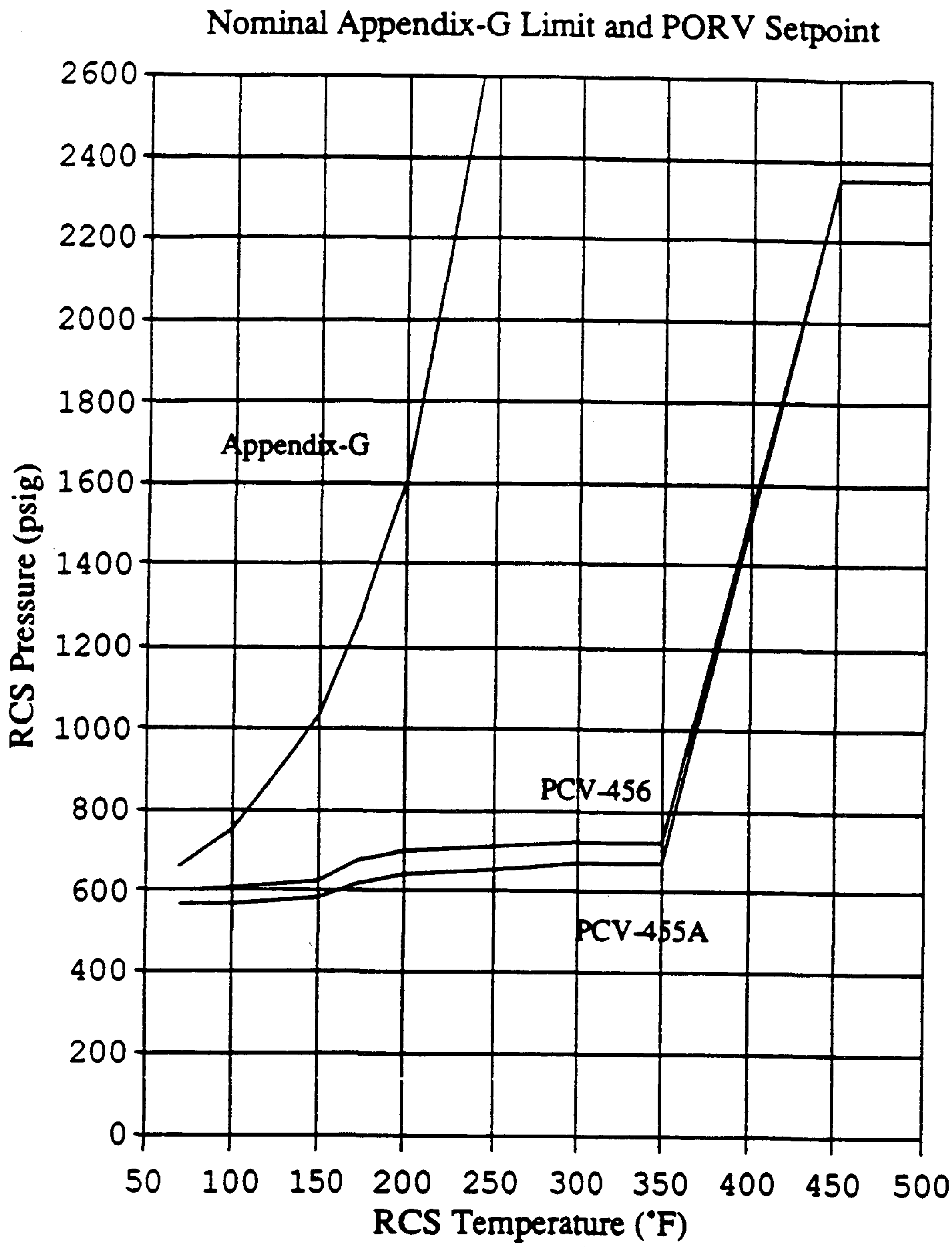


Figure 3.6-2

4.0 PROTECTION SYSTEM SETPOINT METHODOLOGY AND TECHNICAL SPECIFICATION REVISIONS DISCUSSION

Westinghouse performed calculations to determine new instrument uncertainties for most of the protection functions (both RPS and ESF) for the Watts Bar plants. This was necessary to reflect changes due to; removal of the Foxboro process racks and installation of the Eagle-21 process racks, and the addition of the Trip Time Delay (TTD) on Steam Generator Water Level - Low-Low. Removal of the "Old" Steambreak Protection System and replacement with the "New" version resulted in the elimination of several protection functions, and addition of one new function. The effects of these changes are noted on the following pages.

4.1 Effects of Use of Eagle-21 Process Racks

Removal of the Foxboro process racks and installation of the Eagle-21 process racks resulted in at least a small change for each protection function affected. The NIS cabinets were not affected by this change, thus NIS Power Range High Setpoint, NIS Power Range Low Setpoint, NIS Power Range Positive Rate - High and NIS Power Range Negative Rate - High Nominal Trip Setpoints and Allowable Values remain unchanged from those contained in the Proof and Review Technical Specifications. Undervoltage RCP and Underfrequency RCP Reactor Trips do not feed the process racks, thus their setpoints and uncertainties are also unaffected. All remaining protection functions were affected for several reasons; 1) the calibration accuracy (RCA), drift (RD) and temperature effect (RTE) values are significantly different from the Foxboro process racks, 2) there is no physical comparator (RCSA), now performed in the microprocessor, thus that term has been eliminated. The changes in these values have a significant impact on the magnitude of the "T" value and thus the determination of the Allowable Value. Even though there is a reduction in the magnitude of the "T" value, it is still sufficient to encompass the expected drift of the digital process racks.

The most significant change, from a surveillance point of view, is the addition of testing an inoperable channel in "Bypass". The Eagle-21 process racks allow bypassing an inoperable channel when performing surveillance tests on an operable channel. Placing the inoperable channel in "Bypass" results in indication to the operator and allows placing an operable channel in the "Test" mode (which results in it being placed in "Trip"). The applicable "Required Action" statements in the Watts Bar Technical Specifications (Sections 3.3.1 and 3.3.2) have been modified by a Note to reflect this capability.

4.2 Effects of "New" Steambreak Protection System

The "New" Steambreak Protection System is somewhat of a misnomer. In reality the "New" part is the reliance on Steamline Pressure - Low for SI and Steamline Isolation actuation by itself, rather than on the coincidence of Steam Flow in Two Steamlines - High with either Steamline Pressure or Tavg. There is also the deletion of Differential Pressure in Two Steamlines - High and the addition of Negative Steamline Pressure Rate - High (for when Steamline Pressure - Low SI is blocked in Mode 3). Finally, as an adjunct, Steam Flow / Feedwater Flow Mismatch coincident with Steam Generator Water Level - Low Reactor Trip is eliminated. The Nominal Trip Setpoint for Steamline Pressure - Low was not changed as part of this effort. The identified trip functions were eliminated and a Nominal Trip Setpoint was defined for the new actuation function (Negative Steamline Pressure Rate - High). The new trip setpoint is identified in the Proof and Review Technical Specifications and Table 3-21 of WCAP-12096 Revision 5.

4.3 Effects of TTD

The TTD function results in a delay in actuation of a Steam Generator Water Level - Low-Low reactor trip when the power level is less than 50 % RTP. Two algorithms are utilized, one based on only one Steam Generator indicating Low-Low level and the other based on multiple Steam Generators indicating Low-Low level. These algorithms calculate a delay in reactor trip actuation based on the reactor power at the time the logic for trip was fulfilled. These are continuous functions over the power range of 0 to 50 % RTP. This is possible due to the use of the digital process racks. Use of the analog process racks generally results in discrete, stepwise time delays as a function of power level.

The Watts Bar Technical Specifications and WCAP-12096 Revision 5 were modified to reflect the addition of the TTD function. The TTD delays reflect single or multiple affected Steam Generators and a power range of 0 to 50 % RTP.

4.4 Uncertainty Calculations

The uncertainty calculations reflecting the effects noted in the preceding sections are explicitly noted in WCAP-12096, Revision 5 for the following functions:

Overtemperature delta-T (Table 3-5)
Overpower delta-T (Table 3-6)
Pressurizer Pressure - Low reactor trip (Table 3-7)
Pressurizer Pressure - High (Table 3-7)
Pressurizer Level - High (Table 3-8)
Loss of Flow (Table 3-9)
Steam Generator Water Level - Low-Low (Table 3-10)
Containment Pressure - High (Table 3-13)
Containment Pressure - High-High (Table 3-13)
Pressurizer Pressure - Low SI (Table 3-14)
Steamline Pressure - Low (Table 3-15)
Steam Generator Water Level-High-High (Table 3-16)
Negative Steamline Pressure Rate-High (Table 3-17)
RWST Level-Low-Low (Table 3-18)
Containment Sump Level - High/Auto Switchover (Table 3-19)
Vessel ΔT Equivalent to Power (Table 3-20)

5.0 SUPPLEMENTAL DOCUMENTS

1. Functional Requirements (WAT/WBT/300).

2. FSAR Chapter 7 Mark-ups.

3. FSAR Chapter 15 Mark-ups.

4. Technical Specification Mark-ups.

5. Safety Evaluation (SECL-92-194).

6. Environmental and Seismic Qualification, WCAP-8687, Supplement 2-E69A and E-69B (Proprietary), and WCAP-8587, Supplement 1, EQDP-ESE-69 (Non-Proprietary).

7. Noise, Fault, Surge Withstand Capability report, the EMI and RFI Qualification Testing report, WCAP-11733 (Proprietary), and WCAP-11896 (Non-Proprietary).

8. Median Signal Selector (MSS), WCAP-12417 (Proprietary), and WCAP-12418 (Non-Proprietary).

9. Setpoint Methodology, WCAP-12096, Revision 5, (Proprietary), and WCAP-13721 (Non-Proprietary).

10. Eagle Topical Report, WCAP-12374, Revision 1 (Proprietary), and WCAP-12375, Revision 1 (Non-Proprietary).

11. Verification and Validation Report.

12. Field Change Notices:

WATM-10795	EAGLE 21
WATM-10796 A, B, C	NSLB
WATM-10797	MSS
WATM-10800 A, B	SSPS
WATM-10804	EAGLE 21
WATM-10808	EAGLE 21 FRONT TEST PANEL

APPENDIX A

Steam Generator Reference Leg Uninsulated Effect

This is in response to your approach of "having a feedwater line break analysis (by Westinghouse) based on a reactor trip being generated from low-low steam generator level and showing (by TVA) that the containment pressure signal would occur earlier" and your request "that Westinghouse analyses and supporting documentation for this analysis and Eagle 21 continue to be based on a reactor trip associated with the receipt of a low-low steam generator level signal without additional instrument error due to post-accident reference leg heat-up", which "is due to the fact that TVA is showing that taking credit for high containment pressure keeps the current analysis conservative".

Our response is based upon having a clear definition of the boundaries and the "integration" or "overlaying" of the TVA analysis with the Westinghouse analysis that would be maintained from this day forward as long as the SG reference legs are not insulated. The Westinghouse provided documents must be revised based on this clear definition and have an auditable QA trail. This definition is essentially based on a clarification of some pertinent details of the FLB Analysis and the Setpoint Methodology WCAP-12096 and appropriately reflecting the plant "as built" configuration. This clarification and definition are as follows:

Clarification:

A. FLB Analysis Pertinent Details Clarification: The non-LOCA Feedline Break (FLB) Analysis basically considers inside containment and outside containment breaks, both of which are analyzed using 0% span for the SG Low-Low Level setpoint for a reactor trip for protection purposes. For this analysis, the inside containment break case is the worst case, involves adverse environment conditions, and bounds the outside containment break case, which does not encounter adverse conditions. The bounding worst case results are typically reported in the Westinghouse documentation that is transmitted to the utilities. Therefore, the FLB Analysis would be split into two parts: the inside containment analysis part seeing adverse conditions (Reference Leg heatup would be a consideration), and the outside containment analysis part which does not see adverse conditions (Reference Leg heatup would not be a consideration).

B. Setpoint Methodology WCAP-12096 Clarification: The SG Low-Low Level Setpoint (12.0% + 5.0% margin = 17.0% of span to reflect the Technical Specifications) calculated in the Setpoint Methodology WCAP is intended to be a bounding value (normally inside containment assumptions bound outside containment assumptions) and includes adverse environment conditions, which includes an insulated Reference Leg Heat-up error of 3% supplied by TVA. With no insulation on the Reference Leg, this 3% error value would constitute an "unverified assumption", i.e., an assumption made that is recognized as not reflecting the plant "as built" configuration but is considered by TVA to be sufficiently conservative. An adjustment for the non-insulated Reference Leg would increase the SG Low-Low Level setpoint to a value on the order of 24% to 27% of span (with no margin); however, as requested by TVA, this adjustment would not be made. Rather, two setpoints would be provided: one setpoint for the FLB Inside Containment analysis (supported by TVA), and one setpoint for the FLB Outside Containment analysis (supported by Westinghouse).

Definition:

A. TVA would have overall responsibility for the Full Power Feedline Break (FLB) Inside Containment analysis (demonstrating that the SG Low-Low Level Setpoint is not required for a reactor trip), and, for the Part Power Feedline Break Inside Containment analysis (demonstrating the SG Low-Low Level Setpoint is not needed for the TTD Setpoints). TVA would have the responsibility for showing the "integration" or the "overlaying" of the TVA analysis with the Westinghouse analysis.

1. Full Power Feedline Break Analysis:

a. Westinghouse Analysis: The Westinghouse Feedline Break (FLB) Analysis is based upon demonstrating conformance with IEEE-279 requirements and uses the SG Low-Low Level Setpoint, which is a direct measure of the desired variable, for a reactor trip. (All Westinghouse FSAR Chapters 6 & 15 accident analyses are based on demonstrating conformance with IEEE-279 requirements, and use direct measurement of the desired variables for trip setpoints). For this analysis, the SG Low-Low Level trip setpoint was based upon an Insulated SG Reference Leg Error provided by TVA. With no insulation on the reference leg, this function would not actuate with the current setpoint and the FLB analysis would not reflect the "as built" plant configuration. Westinghouse would provide a FLB Inside Containment analysis based on using a SG Low-Low Level setpoint defined by TVA, e.g., the current draft value of 17% span, utilizing Transmitter EA and Reference Leg errors defined by TVA. TVA must support applying this analysis to the plant configuration by "integrating" or "overlaying" the TVA analysis that uses the Containment Pressure High setpoint.

b. TVA Analysis: TVA performed an analysis to demonstrate that the Containment Pressure High trip setpoint would be reached before the SG Low-Low Level trip setpoint for <u>all</u> cases that the Westinghouse Inside Containment analysis supports. TVA would support this analysis and demonstrate that the SG Low-Low Level setpoint is not physically needed. TVA must support applying this analysis to the plant configuration and to the "integrating" or "overlaying" of the Westinghouse analysis that uses a SG Low-Low Level trip setpoint.

2. Part Power Feedline Break Inside Containment Analysis For TTD Setpoints: The preceding also applies for this analysis. For information, the following is noted:

a. Currently, each 1/N and 2/N TTD setpoint is one setpoint for both normal and adverse environment conditions. (With an uninsulated SG Reference Leg, these setpoints would only be applicable to normal conditions.)

b. The 2/N normal condition TTD setpoint is currently based on the results of the Part Power FLB Inside Containment analysis. (The 1/N normal condition setpoints are currently based on the Part Power Loss Of Normal Feed [LONF] analysis.)

B. TVA would have overall responsibility for addressing the Full Power Feedline Break Inside Containment analysis and the Part Power FLB Inside Containment For TTD Setpoints for future analyses, such as Phase 2 Of Upflow Conversion, and future safety evaluations, as applicable, and for addressing any reduction in the minimum AFW MDAFWP flow below 410 gpm. Any flow below this value would also affect the SLB M/E Outside Containment analysis, including WCAP-13274, the Full Power and Part Power LONF analysis, and the Full Power FLB Outside Containment analysis.

C. TVA would be responsible for the Instrumentation Hardware (Transmitter, Control Room Annunciators / Alarms) associated with not getting a trip on SG Low-Low level during a FLB accident.

The Westinghouse documents that would be affected are listed below. These documents (and their corresponding calculation notes) must be revised to reflect the preceding. In order for TVA to "integrate" these analyses, and to maintain a definitive paper trail, the Westinghouse documentation will be revised to reflect a FLB Inside Containment analysis (which utilizes a SG Low-Low Level trip setpoint defined by TVA) and a FLB Outside Containment analysis (which does not consider Reference Leg Heatup). The expected areas of revision are briefly identified. Additionally, appropriate notes to reflect the preceding will be added to this documentation. These notes would reference this letter.

The note would be along the lines of the following: "In accordance with the Westinghouse letter to be issued and TVA letter W-7228 (TVA letter defining SG Low-Low Level trip setpoint), TVA has overall responsibility for the Full And Part Power Feedline Break Inside Containment analyses. The reason is due to TVA's decision not to insulate the SG Reference Legs. TVA has performed an analysis demonstrating that the Containment Pressure High setpoint is reached before the SG Low-Low-Level setpoint defined by TVA is reached in the Westinghouse analysis. In order for TVA to 'integrate' these analyses, and to maintain a definitive paper trail, the Westinghouse documentation will be revised to reflect a FLB Inside Containment Analysis (which utilizes a SG Low-Low Level trip setpoint defined by TVA) and a FLB Outside Containment Analysis (which does not consider Reference Leg Heatup). Therefore, for the current analysis, and future analyses, or safety evaluations, involving the FLB, Westinghouse will provide input to TVA based upon Westinghouse methodology addressing Outside Containment environmental conditions; TVA will complete the analyses, and safety evaluations, addressing Inside Containment conditions."

The documents to be revised include:

A. Eagle 21 Upgrade Program:

1. Functional Requirements: Document 7 for TTD.

2. PL&S: Page 21, Reactor Protection System, 2. Reactor Trips, G. Low-Low SG water level. And, the alarms section.

3. Setpoint Methodology (WCAP-12096, Rev.5, WCAP-13721): Tables 3-10, 3-31 and 4-2 for the trip setpoint.

4. Technical Specifications: Tables 3.3.1-1, 3.3.1-2, for trip setpoints.

5. Safety Evaluation (SECL 92-194, Rev 1): Summary, I&C section (for IEEE-279 criteria), and Non-LOCA section.

6. Summary Report (WCAP-13462): Introduction, Non-LOCA analysis section, I&C Protection section (for IEEE-279 criteria), and the Setpoint section. A copy of this letter, along with a copy of TVA letters W-7176 and W-7228 would be added as an appendix.

7. FSAR Mark-ups: FLB (TVA would add information for using Containment Pressure High trip setpoint).

8. FLB AAPC: This is the Non-LOCA accident information.

9. SSDs (revisions by TVA):
   a. Narrow Range SG Level
   b. Containment Pressure

10. Functional Diversity Assessment (WCAP-13869): FLB