WESTINGHOUSE CLASS 3 (NON-PROPRIETARY)

WCAP-13869
REV. 1

T33 931102 951

FUNCTIONAL DIVERSITY ASSESSMENT
FOR THE
REACTOR PROTECTION SYSTEM/ENGINEERED SAFETY FEATURES
ACTUATION SYSTEM
AT
WATTS BAR UNITS 1 AND 2

OCTOBER 1993

W. J. Scherder

PROJECT: WBN DATE NOV 02 1993
CONTRACT: 71C62--54114--1 FILE: N3M--2-- 2X
Doc. ~~DRAWING~~ NO. WCAP-13869
SHEET NA REV. 1 UNIT 1&2
ISSUED FOR ECN NA

APPROVED

This approval does not relieve the Contractor from any part of his responsibility for the correctness of design, details and dimensions

Date NOV 02 1993

TENNESSEE VALLEY AUTHORITY
enc'd by ltr # W-7177
Watts Bar Project Engineer

Westinghouse Electric Corporation
Nuclear and Advanced Technology Division
P.O. Box 355
Pittsburgh, Pa. 15230

ENCLOSURE 1

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
GENERIC LETTER 88-20 - INDIVIDUAL PLANT EXAMINATION (IPE)
REQUEST FOR ADDITIONAL INFORMATION (RAI) RESPONSE

TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF TABLES

ACKNOWLEDGEMENTS

The authors would like to acknowledge the expertise provided by J. Ackers, J. Bass, L. Erin, R. Sterdis, L. Tomasic, C. Thompson, N. Lewis, and C. Treleani.

PROTECTION SYSTEM DIVERSITY ASSESSMENT

1.0 INTRODUCTION

The purpose of the following discussion is to demonstrate that in the unlikely event of a common mode failure of the Eagle 21 Process Protection System, coincident with a transient analyzed as part of the Watts Bar Units 1 and 2 licensing basis, sufficient diverse means of mitigating the transient are available to bring the reactor to a safe shutdown condition. No attempt is made to demonstrate that the existing safety analysis acceptance criteria would continue to be met since the contemplated scenario is extremely unlikely and beyond the design basis of the Watts Bar units. However, it should be noted that in most cases, if an accident were to occur, the plant initial conditions would be less severe than those analyzed for the FSAR.

Primary and backup protection system signals are provided for each transient comprising the Watts Bar licensing basis. For the purpose of this discussion, a primary protection signal is one upon which the protection function occurs in the licensing basis analysis. Backup protection signals are those expected to occur if the primary signal did not occur. A primary or backup protection signal, derived through a system other than the Eagle 21 process equipment, is available for most transients. Additionally, the ATWS Mitigation System Actuation Circuitry (AMSAC)[1] system is available to provide protection against anticipated transients without reactor trip and is independent of Eagle 21. However, several events, in the absence of Eagle 21, may require a minimum level of operator action to provide necessary protection functions. In these cases, backup protection, alarms, and indicators processed independently of Eagle 21 are available to allow the plant operator to bring the plant to a safe shutdown condition using existing Watts Bar procedures. If manual protection system functions are required, they too are independent of Eagle 21 and can be actuated from the control room.

In Sections 2.1 through 2.4, the licensing basis accidents are divided into four categories: 1) licensing basis events that do not require Eagle 21 for primary or backup protection (Section 2.1); 2) events that do not require Eagle 21 for primary protection but assume Eagle 21 protection system signals for backup (Section 2.2); 3) events that require Eagle 21 for primary protection signals but will receive automatic backup protection from systems other than Eagle 21 (Section 3.3); and 4) events that assume Eagle 21 for primary and backup protection signals for some aspect of the automatic protection (Section 3.4).

In most cases, for the events in "Category 3" (defined as Eagle 21 primary/diverse automatic backup), the backup automatic protection signals occur at approximately the same time as the primary protection signals assumed in the safety analysis. To address any implication of a delay in receiving the backup signal, the non-Eagle 21 backup functions were compared to those assumed in WCAP-7306, "Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors, (Reference 1)." This WCAP is a licensing basis commitment for the Watts Bar units, therefore, any

---

[1] The AMSAC system installed at Watts Bar Units 1 and 2 meets the intent of Section 1.2.7 of NUREG 0493 in that different equipment is used to perform safety functions. The "different" equipment used to decrease the vulnerability to common mode failures consists of separate and independent sensors, cables, signal processing electronics, and power supplies. The system installed at Watts Bar was supplied and manufactured by Atomic Energy of Canada. The system contains two programmable logic controllers supplied by Gould Modicon that use computer microchips manufactured by Sharp Electronics of Japan.

conclusion based on information from WCAP-7306 is applicable to the Watts Bar units. For events in "Category 4" (defined as Eagle 21 primary/Eagle 21 backup) discussions are provided which describes the functions/indicators credited for alerting the operator of the transient, the expected time of occurrence, and the safety classification of these functions/indicators.

Table 1 provides a summary of the protection functions processed through Eagle 21. Table 2 provides a summary of the automatic protection signals processed by systems other than Eagle 21 and passive protection functions assumed in the event specific discussions. Table 3 provides the safety classification and a summary of the key non-protection system indicators and alarms credited for the Category 3 and 4 events. Table 4 provides a summary of the events which credit functions processed in Eagle 21 for primary protection but have automatic backup protection provided independently of Eagle 21. Table 5 provides a summary of the events which assume Eagle 21 for primary and backup protection, as well as indicators and alarms provided by systems other than Eagle 21 that are credited in the event specific discussions to alert the operator of the transient. Table 6 contains a list of acronyms and abbreviations.

2.0 EVENT ASSESSMENT

2.1 Licensing Basis Events That Do Not Require Eagle 21 Process Protection Equipment for Primary and Backup Protection Signals

The events discussed in this section are completely unaffected by a common mode failure to the Eagle 21 Process Protection System. Although events in this category may receive Eagle 21 Process Protection System signals, the primary protection system response and a backup protection signal is provided by a system other than Eagle 21.

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

Both the primary and backup protection for the RCCA Bank Withdrawal from a Subcritical Condition event are provided by the Nuclear Instrumentation System (NIS). The NIS is a separate and independent part of the Reactor Protection System that is not processed by the Eagle 21 protection system. For this event, as analyzed for the FSAR, a reactor trip occurs on power range high neutron flux (low setpoint). Backup protection is provided by the source, intermediate, and power range (high setpoint) high neutron flux and power range high neutron flux rate reactor trips. Additional event mitigation is provided by control-permissive rod stops on high intermediate and power range flux levels, which are also processed by the NIS.

FSAR Section 15.2.3 Rod Cluster Control Assembly (RCCA) Misoperation

Events presented in this FSAR section include one or more dropped RCCAs within the same group, a dropped RCCA bank, and a statically misaligned RCCA. Most events within this group do not require a reactor trip for accident mitigation. Some dropped RCCAs may result in a reactor trip on power range negative flux rate. For a dropped RCCA bank with a reactivity worth of greater than $0.5\%\Delta\rho$, an immediate reactor trip on high negative neutron flux rate will occur. Indications of dropped RCCAs or a statically misaligned RCCA are provided by rod deviation alarms, rod position indicators, and asymmetric power distribution as seen by NIS detectors and core exit thermocouples. All of these signals are provided by systems other than Eagle 21.

FSAR Section 15.3.3 Inadvertent Loading of a Fuel Assembly Into an Improper Position

Indication of improper fuel loadings, as discussed in the FSAR, are detected by the Incore Instrumentation System or abnormally high readings on the core exit thermocouples. Both of these indications are processed by systems other than Eagle 21.

FSAR Section 15.4.6 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

For the RCCA Ejection event, both primary and backup protection are provided by the Nuclear Instrumentation System (NIS). As noted previously, the NIS is independent of the Eagle 21 process system. For this event, cases are analyzed at hot zero power and hot full power. Reactor trip occurs on power range high neutron flux - low setpoint or high setpoint depending on the initial reactor power level. For cases initiated from zero power, backup protection is provided by the source, intermediate, and power range (high setpoint) high neutron flux and high positive neutron flux rate reactor trips. For cases initiated from full power, backup protection is provided by the high positive neutron flux rate reactor trip.

2.2 Licensing Basis Events That Do Not Require Eagle 21 Process Protection Equipment for Primary Protection Signals, But Assume Eagle 21 Process Signals for Backup Protection

The analysis of events discussed in this section are completely unaffected by a common mode failure of the Eagle 21 Process Protection System since the primary protection system responses are derived through systems other than Eagle 21 or no protection system response is required for core or reactor coolant system protection. Backup protection signals are provided by the Eagle 21 Process Protection System.

FSAR Section 15.2.6 Startup of an Inactive Reactor Coolant Pump

The FSAR analysis of this event conservatively credits a reactor trip on high neutron flux (power range - high), which is processed by the NIS system. A backup protection signal is provided by the low reactor coolant flow (Eagle 21) when the power increases above the P-8 setpoint. Although adequate protection is provided, since 3 loop operation during startup and power operation is not permitted per Watts Bar Units 1 and Unit 2 Technical Specification 3.4.4, this event is not considered credible for the Watts Bar units.

FSAR Section 15.2.11 Excessive Load Increase Incident

An Excessive Load Increase event is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core and steam generator load demand. The event, as analyzed for the FSAR, assumes a 10% step load increase. No reactor trip is required for event mitigation. The reactor achieves a new equilibrium condition at a higher power level corresponding to the increased steam flow at conditions which do not challenge any applicable safety limits. Protection, if required, is provided by overtemperature ΔT (Eagle 21), overpower ΔT (Eagle 21), and power range high neutron flux (NIS) reactor trips.

FSAR Section 15.2.14 Inadvertent Operation of Emergency Core Cooling System

No reactor trip is required for core protection. However, for the event as analyzed in the FSAR, a spurious safety injection (SI) signal will eventually cause a reactor trip to occur on low pressurizer pressure (Eagle 21). The event is detected by indications other than those provided by Eagle 21. These include safety injection pump start and discharge flow indicators and high negative neutron flux rate signal (NIS). If the event were to proceed for an extended length of time, further indication would be provided by the pressurizer safety relief valve (PSRV) position indication (assuming the PORVs do not function due to common mode failure of Eagle 21).

FSAR Section 15.3.4 Complete Loss of Forced Reactor Coolant Flow

Primary reactor protection for a Complete Loss of Forced Reactor Coolant Flow event, resulting from a simultaneous loss of electrical power to all reactor coolant pumps, is provided by reactor trip on a reactor coolant pump undervoltage signal. For the related underfrequency event, primary protection is provided by reactor trip on reactor coolant pump underfrequency. Backup protection is provided by reactor trip on low reactor coolant flow (Eagle 21) and overtemperature ΔT (Eagle 21.)

2.3 Licensing Basis Events That Require Eagle 21 Process Protection Equipment for Primary Protection Signals, But Will Receive Backup Automatic Protection Signals from Systems Other than Eagle 21 - (Category 3 Events, Eagle 21 Primary Protection/Diverse Automatic Backup)

The events discussed in this section receive primary protection system signals from the Eagle 21 Process Protection System and would be affected by a common mode failure to the Eagle 21 Process Protection System. However, backup protection signals are available that would automatically provide the necessary protection functions through systems other than the Eagle 21 Process Protection System.

With the exception of a Single RCCA Withdrawal and Feedline Break event, all events in this category are classified as ANS Condition II events and have been analyzed by Westinghouse in WCAP-8330, Westinghouse Anticipated Transients Without Trip Analysis," Reference 2. Above C-20 (40% rated thermal power, RTP), the AMSAC system is available to provide the necessary protection functions. The AMSAC system initiates auxiliary feedwater and trips the turbine. Above P-9 (50% RTP) the transients would be less severe than postulated for ATWS events since an automatic reactor trip on turbine trip will occur independent of the Eagle 21 Process Protection System. Below C-20, generic analyses applicable to Watts Bar performed for ATWS events have demonstrated that the AMSAC system is not required to prevent reactor coolant system damage. Additionally, with decreasing reactor power, the operator has a much greater time to manually initiate any required protection functions. A discussion of each event is provided in this section.

FSAR Section 15.2.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power
FSAR Section 15.3.6 Single Rod Cluster Control Assembly Withdrawal at Full Power

For Uncontrolled RCCA Withdrawal at Power events, depending on the reactivity insertion rate, reactor protection is provided by the Nuclear Instrumentation System (NIS power range high neutron flux - high setpoint) or Eagle 21 (overtemperature ΔT reactor trip). In the event of a common mode failure to the Eagle 21 Protection System, backup protection would be provided by the power range high neutron flux (high setpoint) reactor trip. Similar to the RCCA Withdrawal from Subcritical event, backup protection is also provided by high positive neutron flux rate (NIS) reactor trip and control-permissive rod stops on high power range flux level (NIS). Additional indication of the event is provided to the operator by the high neutron flux alarm and indicator (NIS) (which occurs before the high neutron flux reactor trip signal) and by rod position indicator and alarms. Passive overpressure protection is provided by the pressurizer and steam generator safety valves.

Generic Uncontrolled Bank Withdrawal at Power analyses performed for WCAP-7306 demonstrated that the availability of the high neutron flux power range reactor trip alone would prevent the minimum departure from nucleate boiling ratio (DNBR) in the core from decreasing below 1.0. A DNBR of 1.0 was used in WCAP-7306 as an acceptable limit for Condition II transients with a failure of the primary protection signal. However, since only rod withdrawals resulting in slow reactivity excursions require a reactor trip signal from the Eagle 21 Process Protection System for primary protection, most failures which would cause a spurious rod withdrawal are alarmed to the operator via the NIS such that the operator would trip the reactor prior to the time an automatic reactor trip is required.

FSAR Section 15.2.4 Uncontrolled Boron Dilution

For most Uncontrolled Boron Dilution scenarios the primary protection system signals assumed in the analysis are processed outside Eagle 21. For a Boron Dilution event during power operation, with the reactor in manual rod control mode, indication of the event is provided to the operator by the overtemperature ΔT reactor trip, which is processed by Eagle 21. Backup indication is provided by the high power range neutron flux (NIS) reactor trip. When in automatic rod control mode, event detection is provided by the low rod position and the low-low rod position alarms, neither of which are processed by Eagle 21.

During reactor startup, indication of an Uncontrolled Boron Dilution event is provided by a high source range neutron flux reactor trip signal. During hot shutdown and cold shutdown conditions indication of a boron dilution event is provided by a high power range neutron flux rate or high source range flux level alarms. Both of these alarms are processed by the NIS. Although a Boron Dilution accident is administratively precluded from occurring during refueling, indication of the event would also occur on a high source range count rate alarm or high source range flux level alarm (both NIS).

FSAR Section 15.2.7 Loss of External Electrical Load and/or Turbine Trip

For a complete loss of electrical load, resulting from a turbine trip, primary protection is assumed to occur (depending on initial operating conditions) on high pressurizer pressure, overtemperature ΔT, or low-low steam generator water level. All of these protection function signals are provided by Eagle 21. However, in the case of a turbine trip, above 50% RTP (P-9) backup protection is provided by reactor trip on turbine trip, which is processed outside of the Eagle 21 protection system.

Generic analyses of the loss of external electrical load event without subsequent turbine trip, assuming no automatic steam dump or reactor trip, were performed for WCAP-7306 and WCAP-8330 (ATWS analyses). In the analysis performed for ATWS the minimum DNBR calculated was above the safety analysis limit. Both analyses determined that an automatic reactor trip was not required to meet over pressure limits. The pressurizer and steam generator safety valves are sized to protect the RCS and steam generator against overpressurization for all load losses without assuming operation of the steam dumps, pressurizer spray, power operated relief valves, or automatic rod control. The steam generator safety valve relief capacity is sized to remove the steam flow at the Engineered Safeguards Design rating ($\sim 105\%$ of the steam flow at rated power) from the steam generator without exceeding 110% of the steam system design pressure. The pressurizer safety valve capacity is sized for a complete loss of heat sink with the plant initially operating at the maximum calculated design turbine load and with operation of the steam generator safety valves. The pressurizer safety valves are then able to maintain system pressures within 110% of the RCS design pressures without direct or immediate reactor trip.

FSAR Section 15.2.8 Loss of Normal Feedwater

Primary protection (reactor trip and auxiliary feedwater initiation) is provided for this event by a low-low steam generator water level signal, which is processed by Eagle 21. Backup protection is provided by the high pressurizer pressure (assuming no power operated relief valves), high pressurizer water level, or overtemperature ΔT reactor trips. All of the backup protection signals are processed by Eagle 21. In the event of an Eagle 21 common mode failure, backup protection would be provided by the AMSAC system. Above 40% RTP (C-20) the AMSAC system will automatically initiate auxiliary feedwater and trip the turbine on a low steam generator water level signal. Generic ATWS analyses (Reference 2), applicable to Watts Bar, have demonstrated that by tripping the turbine and initiating auxiliary feedwater the AMSAC system will prevent damage to the RCS and core. These same generic analyses have demonstrated that the automatic protection functions initiated by AMSAC are not needed below 40% RTP. It should be noted that the Eagle 21 failure scenario is less severe than postulated for the ATWS scenario since an automatic reactor trip will occur above P-9 (50% RTP) independent of Eagle 21 on turbine trip.

FSAR Section 15.2.9 Loss of Offsite Power to the Station Auxiliaries

The Loss of Offsite Power to Station Auxiliaries event is similar to the Loss of Normal Feedwater event except that power is lost to the reactor coolant pumps upon reactor trip. The event is analyzed to determine the adequacy of the auxiliary feedwater system in conjunction with natural RCS circulation in removing the core decay heat. The event as analyzed for the FSAR assumes the loss of offsite power occurs upon reactor trip on low-low steam generator water level (Eagle 21).

Assuming the loss of offsite power at the time of reactor trip is conservative since this provides the minimum steam generator inventory for removing core decay heat at the reactor trip (mass corresponding to the low-low steam generator water level setpoint). However, should a loss of offsite power to the station auxiliaries occur which resulted in a loss of power to the reactor coolant pumps (RCP), backup reactor trip signals will occur at the initiation of the event on either reactor coolant pump undervoltage, underfrequency, or loss of condenser vacuum and subsequent turbine trip which are independent of Eagle 21. Auxiliary feedwater will automatically be initiated via the AMSAC system or manually initiated following reactor trip via existing Watts Bar procedures.

FSAR Section 15.2.10 Excessive Heat Removal Due to Feedwater System Malfunctions

Feedwater System Malfunctions resulting from excess feedwater flow are analyzed at zero and full power. For the zero power cases, primary and backup protection are provided by the NIS which is separate and independent of Eagle 21. Thus, the analysis performed at hot zero power, to bound lower modes of operation, is not affected by failures of the Eagle 21 system.

The at-power cases analyzed for the FSAR are terminated by feedwater isolation on high-high steam generator water level signal. The high-high steam generator water level signal also trips the turbine which in turn initiates a reactor trip signal. Although the event as analyzed for the FSAR assumes a reactor trip on turbine trip, the overtemperature ΔT (Eagle 21) reactor trip will occur prior to the DNBR limit being exceeded and is considered the primary protection against DNBR for this event. Backup protection is provided from the high power range neutron flux (NIS) and overpower ΔT (Eagle 21) reactor trips. With respect to challenges to the core, this event does not usually require a reactor trip to meet the DNB design basis. At worst, without an automatic reactor trip the core will reach an equilibrium condition with a minimum DNBR above the design limit of 1.0. A DNBR of 1.0 was used in WCAP-7306 as an acceptable limit for Condition II transients with a failure of the primary protection signal.

Feedwater isolation is required to prevent excessive moisture carryover to the turbine and water in the steam pipes (which could cause a steam line break). Automatic feedwater isolation occurs on a high-high steam generator water level signal (Eagle 21). Indication/detection is available to the operator via the steam generator monitoring instrumentation provided through the Auxiliary Feedwater System Level Instrumentation Loops and Feedwater System monitors (Table 3, Item 18).

FSAR Section 15.4.2.2 Major Rupture of a Main Feedwater Pipe

For the Feedline Break event, reactor trip and auxiliary feedwater start are required for accident mitigation. The primary protection signal assumed for a feedline break is low-low steam generator water level (Eagle 21). In the absence of Eagle 21 automatic protection is provided by the AMSAC system. Above 40% RTP (C-20) the AMSAC system will automatically actuate auxiliary feedwater and trip the turbine. Above 50% RTP (P-9) a reactor trip will occur on the turbine trip signal. Additionally, the operator would also receive indication of the event via independent steam generator level monitoring via the Auxiliary Feedwater System Level Instrumentation Loops. Similar to the loss of normal feedwater and station blackout events, the AMSAC and steam generator level low indications through the Auxiliary Feedwater System Level Instrumentation Loops would occur prior to steam generator tube uncovery and significant RCS heatup. Once initiated, the auxiliary feedwater flow will be significantly greater than that assumed in the licensing basis analysis since the analysis assumes a single failure to the auxiliary feedwater system.

Additional indicators and alarms processed by systems other than Eagle 21 which would provide indication of the event include: core exit thermocouples (high reading), pressurizer relief tank indicators (high level, temperature, and pressure), pressurizer safety relief valve position indication, and Feedwater System monitors. Above P-9 these signals are not required since all protection functions required by the safety analysis will occur automatically by the AMSAC system (auxiliary feedwater pump start and reactor trip on turbine trip).

Note that below 40% RTP the automatic protection functions initiated by AMSAC and below 50% RTP reactor trip on turbine trip are disabled. Analysis of feedline breaks at power levels below the C-20 setpoint were performed for the steam generator low-low water level trip time delay (TTD, Reference 3). Results of the analysis demonstrated that as the feedline break size and reactor power is decreased the operator has an increasing amount of time available to manually trip the reactor and initiate auxiliary feedwater prior to violating the conservative acceptance criteria Westinghouse applies to this event and an even greater time to initiate these protection functions prior to exceeding the ANS Condition IV acceptance criteria.

Concluding, it should be noted that a Feedline Break event is classified as an ANS Condition IV event and, in itself, is extremely unlikely independent of a common mode failure to the Eagle 21 Protection System. Furthermore, the Watts Bar units will not typically operate at powers below the P-9 setpoint for an extended period of time. Therefore, for the majority of reactor life, if a design basis feedline break event were to occur coincident with a common mode failure to the Eagle 21 system, it is determined that plant protection would be provided automatically by the AMSAC system. For the remainder of plant operating conditions the operator would have a significant amount of time to manually provide the necessary protection response.

2.4 Licensing Basis Events That Require Eagle 21 Process Protection Equipment for Primary and Backup Protection Signals - (Category 4 Events, Eagle 21 Primary Protection/Eagle 21 Backup)

The events discussed in this section receive both primary and backup protection signals for some aspect of the protection system response assumed in the safety analyses by the Eagle 21 Process Protection System. Alarms, indicator lights, and recorders are available for these events which will provide the operator with indication of an event. Operator action would be required to manually provide some of the necessary protection system functions should a common mode failure occur in the Eagle 21 process protection system. It should be noted that in most cases these alarms/indicators would occur prior to a reactor trip signal or other protection system signal would have been received.

Steam Line Break events are included in this category since operator action would be required for feedwater isolation and safety injection. Backup reactor trip signals for Steam Line Break events occurring at power are provided via the Nuclear Instrumentation System.

Loss of Forced Reactor Coolant Flow

FSAR Section 15.2.5 Partial Loss of Forced Reactor Coolant Flow
FSAR Section 15.4.4 Single Reactor Coolant Pump Locked Rotor

A loss of forced reactor coolant flow to more than one loop would result in a direct reactor trip on RCP undervoltage or underfrequency which are independent of Eagle 21. Primary reactor protection for a Partial Loss of Forced Reactor Coolant Flow event (loss of flow in less than 4 loops) is provided by low reactor coolant flow signals (2 out of 3) in any reactor coolant loop above P-8. Below P-8 a reactor trip is not required for loss of flow in one reactor coolant loop. The low reactor coolant flow signals are provided by the Eagle 21 system. For a 1 out of 4 loop partial loss of flow event a direct reactor trip may not occur for all core reactivity conditions.

Indications of a 1 out of 4 loop Partial Loss of Flow event would be similar to the Locked Rotor event. However, since the reactor coolant pumps have high inertia flywheels, the length of time for the flow to decrease would be significantly longer for a one-loop Partial Loss of Flow event than it would be for a Locked Rotor event. Indications of a one-loop Partial Loss of Flow and Locked Rotor event include reactor coolant pump breaker position open (alarm and indicator light), reactor coolant pump overcurrent trip, and abnormal pump seal flow indications (see Table 3, Item 10). Other event indications, not directly related to the pump failure, are the pressurizer safety relief valve position indication system when the pressurizer power operated relief or safety valves open, core exit thermocouples (high reading), and steam generator water level low indication provided by the Auxiliary Feedwater System Instrumentation Loops. All indications related to the failed pump occur at the initiation of the event. The other event indications occur prior to the normal event specific acceptance criteria being violated. All indications discussed are provided independent of Eagle 21.

As previously stated, the partial loss of flow event without Eagle 21 would be similar but far less severe than the Locked Rotor event. Watts Bar specific analysis has shown that an immediate reactor trip is not necessary for the Locked Rotor event to meet Condition IV acceptance criteria (no fuel melt, zirconium-water reaction less than 16%). At reactivity conditions corresponding to the beginning of the fuel cycle analysis demonstrated that the operator had approximately 5 minutes to manually trip the reactor and still meet design basis Condition IV acceptance criteria. As the fuel cycle progresses and the moderator temperature coefficient becomes more negative the amount of time available for operator action increases. In particular it should be noted that the pressurizer safety valves were sufficient to maintain the reactor coolant system pressure below 110% of the design pressure. REV. 1

FSAR Section 15.2.12 Accidental Depressurization of the Reactor Coolant System

An Accidental Depressurization of the RCS could occur as the result of an inadvertent opening of a pressurizer relief or safety valve. Primary and backup protection is provided by a reactor trip on a low pressurizer pressure or overtemperature ΔT signal. Both of these reactor trips are processed by the Eagle 21 process system. If the Eagle 21 process system failed, an automatic reactor trip would not occur for this event. Signals processed outside the Eagle 21 process system that would provide the operator with indication of an event are wide range containment pressure indicators and recorder, pressurizer safety or relief valve position indication, pressurizer and safety valve discharge temperature (high reading), PSRV position indication system alarms, and pressurizer relief tank (level, temperature, and pressure) indicators.

Generic analysis of this event without reactor trip for ATWS events (WCAP-8330) showed the event not to be limiting, with the DNB design basis met. Should the event proceed long enough for hot leg saturation to occur, the operator would have had many other indications and sufficient time to mitigate the transient.

<u>Loss of Coolant Accidents - (Small and Large Break LOCA)</u>

The discussion provided below demonstrates that even if automatic protection functions fail, diverse signals are available which would prompt operator action to mitigate the event and bring the plant to a safe shutdown condition. The licensing basis small break LOCA is classified as an ANS Condition III event, while the large break LOCA is classified as an ANS Condition IV event. A LOCA occurring coincident with a common mode failure to Eagle 21 is considered extremely unlikely and beyond the design basis of the Watts Bar units.

For a loss of coolant accident (LOCA), assuming a total failure of Eagle 21, operator action would be necessary to initiate reactor trip and engineered safety feature actuation (safety injection and containment spray). As discussed below, there are many diverse indications available that are processed outside of Eagle 21 which would rapidly provide the operator with indication of a LOCA. Existing Watts Bar procedures would allow for diagnosis of the event and manual response from the control room, independent of Eagle 21. The primary indicators and alarms which would provide the operator with indication of a LOCA as well as the safety classification are given in Tables 3 and 5.

FSAR Section 15.3.1 Loss of Reactor Coolant from Small Rupture Pipes or from Cracks in Large Pipes that Actuate Emergency Core Cooling System (Small Break LOCA)

FSAR Section 15.4.1 Major Reactor Coolant System Pipe Ruptures (Large Break LOCA)

Depending on break size, primary protection (reactor trip, containment spray and safety injection) for a LOCA is provided by low pressurizer pressure signal or high containment pressure signals. All of these protection signals are processed by Eagle 21. For a large break LOCA, no credit is taken for the negative reactivity insertion of the control rods. With or without reactor trip during a LOCA (large or small), core reactivity would automatically decrease due to void formation in the reactor coolant. For large and some small break LOCA scenarios, this would be sufficient to completely shutdown core reactivity. Additional coolant will be automatically provided by the accumulators (when the RCS pressure drops below accumulator injection pressure) and by increased makeup flow (prior to SI initiation) due to the decreased RCS pressure. Containment vent isolation will occur automatically, independent of Eagle 21 on high containment radiation signals. Containment pressure and temperature are passively controlled by the Ice Condenser System.

There are many signals provided by systems other than Eagle 21 that would provide the operator with rapid indication of a LOCA and, based on existing Watts Bar procedures, allow the operator to place the reactor in a safe shutdown condition. A partial list of indicators and alarms include: high containment radiation signals (upper compartment, lower compartment, containment purge air exhaust (gas), and shield building vent monitor), adverse containment environment (moisture, temperature, pressure - high), ice condenser inlet door (open), core exit thermocouples (high), RCP pump seal injection filter high ΔP, and reactor building floor and equipment sump level (high). For larger small breaks and large break LOCAs the operator would observe reactor coolant pump amperage anomalies due to RCP cavitation. The resulting amperage fluctuations could be severe enough to trip the pumps and generate an automatic reactor trip signal through functions not processed by Eagle 21.

The exact time that indicators and alarms are actuated depends on the RCS break size. Based on Watts Bar specific simulator runs for a 4-inch diameter and a large double-ended rupture, the indicators and alarms occur within 30 seconds and prior to completion of accumulator injection. These signals in conjunction with existing Watts Bar procedures would result in operator verification of the event and a manual trip of the reactor. Watts Bar Emergency Operating Procedures would then provide for initiating safety injection and a subsequent controlled cooldown of the RCS. All manual actuations are easily performed in the control room and not subject to common mode failure to Eagle 21.

It should be noted that passive safety injection for most LOCA scenarios would be provided by the accumulators, which do not require any automatic or manual action. Also, makeup flow will increase prior to safety injection due to decreased RCS pressure, which is not assumed in the FSAR analyses, and would provide additional mass for cooling. For very small breaks, the charging pump makeup flow is adequate to sustain pressurizer level. Once safety injection is initiated the flow rate would be significantly greater than that assumed in the licensing basis LOCA analyses. The licensing basis analyses assume only one train of safety injection. As mentioned previously, prior to manual reactor trip, core power would automatically decrease due to void formation and the decrease in moderator density.

Generic analyses have been performed to investigate the effect of delays in initiation of safety injection on the licensing basis large break LOCA (double-ended rupture). NSAC 130 "The Effect of Diesel Start Time Delay on Westinghouse PWRs," presents the results of LOCA analyses in which safety injection is delayed until after the accumulators have injected. The analyses suggest that if the operator initiated safety injection before or shortly after (within 20 seconds) complete accumulator injection the licensing basis acceptance criteria could be met. It should be noted that the analyses presented in NSAC 130 used the standard conservative modelling requirements of 10 CFR 50 Appendix K. For smaller breaks, due to the slower rate of RCS depressurization, the operator would have additional time to initiate safety injection and still bring the reactor to a safe shutdown condition.

Information on the probability of LOCAs is provided in the Watts Bar Probabilistic Risk Assessment (PRA). The PRA classifies breaks from $\geq$ 3.14 in^2 to $\geq$ 28.27 in^2 as medium break LOCAs with a probability of 4.65E-4 per reactor year. Breaks larger than 28.27 in^2 are considered large LOCAs and have a probability of 2.03E-4 per reactor year. For the more probable, but still unlikely smaller break LOCAs (up to 3.14 in^2), the success criteria used for the Watts Bar PRA program are such that if the operator initiates manual safety injection (and any other ESF functions as specified by the emergency operating procedures) within 10 minutes, core damage is precluded. For a larger break, the operator would have less time to prevent core damage, but would still have time to prevent containment bypass or failure, thereby isolating the outside environment from potential radiation releases.

Steam Line Break Events

FSAR Section 15.2.13 Accidental Depressurization of the Main Steam System
FSAR Section 15.4.2.1 Rupture of a Main Steam Line

Steam Line Break Events Analyzed At-Power for Core Response (RTD Bypass Elimination), Containment Analysis (FSAR 6.2.1.3.10), Outside Containment Equipment Qualification

Steam line break events are analyzed for Watts Bar with the reactor at-power and post-tripped with the most reactive control rod stuck out of the core at the onset of the event. Reactor trip (at-power cases), safety injection and feedwater isolation are required for event mitigation. Several reactor trip signals, from systems other than Eagle 21, are available as backup signals. These include the high neutron flux (all ranges, depending on initial power level) and high power range neutron positive flux rate reactor trips (all NIS). Borated coolant will be automatically provided by the accumulators if the RCS pressure drops below the accumulator injection pressure.

In the absence of Eagle 21, actuation of safety injection will require operator action. Watts Bar Emergency Operating Procedures provide the necessary guidance following a reactor trip to ensure that operator action will be taken to confirm and manually initiate any engineered safeguard actuation required to mitigate the steam line break and bring the reactor to a safe shutdown condition. With respect to safety injection, due to conservative modeling assumptions such as minimum temperature, boron concentration, and flow (analyses assume a failure to a safety injection train), a delay in safety injection actuation is not critical to meeting the applicable acceptance criteria.

FSAR Section 15.4.3 Steam Generator Tube Rupture (SGTR)

Primary protection for this event is provided by a reactor trip on overtemperature ΔT. Backup protection is provided by the low pressurizer pressure, high steam generator water level turbine trip, and SI signals. All of these protection signals are generated by Eagle 21. Safety injection is initiated via a low pressurizer pressure signal, but is not required for core protection. Signals generated by systems other than Eagle 21 are the main steam line, condenser vacuum, and steam generator blowdown radiation indicators and alarms. The operator would also notice a decrease in the volume control tank level.

The safety analysis of the SGTR event does not assume normal RCS makeup as a source of reactor coolant since this is conservative. For the Watts Bar units which are four loop plants the more credible scenario would be for the RCS charging system to maintain the primary inventory sufficiently to delay an automatic reactor trip signal, with the operator's first indication of an SGTR event being a steam line, steam generator blowdown, or condenser vacuum radiation monitors. All of these radiation monitors are diverse, and equipped with independent monitors and annunciators and would provide multiple indications of the event. Upon annunciation of any of these signals, existing Watts Bar operating procedures will provide the operator with the guidance necessary to effectively mitigate the SGTR event.

3.0 CONCLUSIONS

The Watts Bar Units 1 and 2 licensing basis accident analyses were reviewed to determine which events required the Eagle 21 Process Protection System for primary or backup protection. Those transients identified as requiring the Eagle 21 Process Protection System for primary protection system response were reviewed to determine if a timely diverse means of automatically mitigating the transient was available or annunciators and indicators were available to allow the operator to diagnose the event and bring the plant to a safe shutdown condition.

For most transients no operator action is required since adequate non-Eagle 21 based automatic functions exist. For several events some operator action is required. In these cases, backup protection functions, alarms, and indicators processed independent of Eagle 21, along with existing Watts Bar operating procedures and Emergency Operating Procedures, are available to bring the plant to a safe shutdown condition.

4.0 REFERENCES

1. WCAP-7306, "Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors," April 1969.

2. WCAP-8330, "Westinghouse Anticipated Transients Without Trip Analysis," August 1974.

3. WCAP-13462, "Summary Report Process Protection System Eagle 21 Upgrade, NSLB, MSS and TTD Implementation Watts Bar Units 1 and 2," June 1993.

Table 1

Eagle 21 Protection System

Eagle 21 Protection Channels

1. Average Temperature and ΔT
2. Pressurizer Pressure
3. Pressurizer Water Level
4. Steam Flow (1)
5. Feedwater Flow (1)
6. Reactor Coolant Flow
7. Turbine Impulse Chamber Pressure
8. Steam Pressure
9. Reactor Coolant Wide Range Temperature (Hot and Cold)
10. Reactor Coolant Wide Range Pressure
11. Steam Generator Narrow Range Water Level
12. Steam Generator Wide Range Water Level
13. Pressurizer Vapor Temperature
14. Pressurizer Liquid Temperature
15. Refueling Water Storage Tank Level
16. RHR Pump Discharge Temperature
17. Boric Acid Tank Level (1)
18. Containment Sump Level
19. Containment Pressure
20. Containment Spray Flow

Eagle 21 Protection Functions Assumed in the FSAR Chapter 15 Safety Analyses

1. Overtemperature ΔT Reactor Trip
2. Overpower ΔT Reactor Trip
3. Low and High Pressurizer Pressure Reactor Trip
4. High-High Pressurizer Level Reactor Trip
5. Steam Line Break Protection
   a. High-High Containment Pressure Steam Line Isolation
   b. Low Steam Line Pressure Steam Line Isolation and Safety Injection
   c. Low Pressurizer Pressure Safety Injection
   d. High Containment Pressure Safety Injection
6. Low Reactor Coolant Flow Reactor Trip
7. Steam Generator Water Level (Low-Low) Reactor Trip and Auxiliary Feedwater
8. Steam Generator Water Level (High-High) Turbine Trip and Feedwater Isolation

Note

(1) These functions do not provide automatic protection.

Table 2

Protection Systems, Monitors, and Indicators Processed Through Equipment Independent of the Eagle 21 Process Protection System

Protection Functions Assumed in the FSAR Chapter 15 Safety Analyses Processed Outside Eagle 21

1. Nuclear Instrumentation System
   a. High Neutron Flux - Source, Intermediate, and Power Range (High & Low) Reactor Trip
   b. High Neutron Flux Rate - Negative and Positive Reactor Trip
   c. High Neutron Flux - Intermediate and Power Range Control Rod Stops
2. Reactor Trip on Turbine Trip
3. Reactor Coolant Pump Undervoltage Reactor Trip
4. Reactor Coolant Pump Underfrequency Reactor Trip

Passive Protection Functions Available

1. Pressurizer Safety Valves
2. Steam Generator Safety Valves
3. Accumulators
4. Ice Condenser System

Table 3
Event Indicators (Page 1 of 5)

Key Alarms and Indicators Processed Outside Eagle 21[1]

1. Core Exit Thermocouple (High and Low) Monitors
2. Radiation Indicators/Recorders
    a. Containment (Upper & Lower Compartment)
    b. Shield Building Vent
    c. Containment Purge Air Exhaust
    d. Steam Line
    e. Condenser Vacuum
    f. Steam Generator Blowdown
3. Nuclear Instrumentation System Flux Detectors
4. High Neutron Flux Alarms (High, Intermediate, and Source Range)
5. Power Range High Neutron Flux Rate Alarm and Light
6. Rod Control System
    a. Rod Position Alarms (Insertion Limit and Deviation)
    b. Rod Bottom Signal Alarm
    c. Rod Position Indicators
7. Steam Generator Level thorough Auxiliary Feedwater System Level Instrumentation
8. Pressurizer Relief Tank
    a. Level Indication
    b. Temperature Indication
    c. Pressure Indication
9. Reactor Coolant Pump Protection
    a. Overcurrent Indication
    b. RCP Breaker Position Abnormal Indicator
    c. Seal Leakoff Flow (high/low) Recorder/Alarm
    d. RCP High Vibration/Loose Parts Monitor & Annunciator
    e. RCP Seal Injection Flow Indicator
    f. RCP Seal ΔP Indicator/Alarm
    g. RCP Seal Outlet Temperature Indication
10. Accumulator Level & Pressure Indication
11. Pressurizer Safety Relief Valve Position Indication System
12. Pressurizer Relief and Safety Valve Discharge Temperature
13. Steam Generator Safety Valve, Relief Valve, and Steam Dump Position Indicator
14. Containment Pressure Monitor

List continues on next page

Notes

(1) Safety classification provided on last three pages of Table 3

Table 3 (continued)
Event Indicators (Page 2 of 5)

15. Ice Condenser System
    a. Door Open Alarm
    b. Ice Temperature Recorder
16. Containment Environment
    a. Moisture
    b. Temperature
17. Safety Injection Pump Status Indication (On/Off) and Discharge Flow Indication
18. Feedwater System
    a. Main Feedwater Pump Speed Monitor
    b. Main Feedwater Pump Status (Running/Tripped)
    c. Main Feedwater Pump Discharge Pressure
    d. Main Feedwater Pump Discharge Flow
19. Reactor Building Floor and Equipment Sump Level Alarm

Table 3 (continued)
Event Indicators (Page 3 of 5)

EVENT INDICATORS	SAFETY CLASSIFICATION	EVENT ASSUMED
Core Exit Thermocouples	RG 1.97 Cat. 1	LOCA (large and small) Steam Line Breaks Locked Rotor Partial Loss of Flow Feedline Break
Radiation:		
a. Containment Upper & Lower Compartment	RG 1.97 Cat. 1	LOCA (large and small) Steam Line Breaks
b. Shield Bldg. Vent (noble gas)	RG 1.97 Cat. 2	LOCA (large and small) Steam Line Breaks
c. Purge Air Exhaust Monitor (gas)	Class 1E	LOCA (large and small) Steam Line Breaks
d. Steam Line	RG 1.97 Cat. 2	SGTR
e. Condenser Vacuum Pump Exhaust	RG 1.97 Cat. 2	SGTR
f. SG Blowdown	Non-Class 1E	SGTR
NIS Indicators	RG 1.97 Cat. 1	Rod Withdrawals at Power
High Neutron Flux Alarms	Non-Class 1E	Steam Line Breaks Rod Withdrawals at Power Boron Dilution
Power Range High Neutron Flux Rate Alarms	Non-Class 1E	Locked Rotor Partial Loss of Flow Steam Line Breaks Rod Withdrawals at Power Boron Dilution
Rod Control System		Rod Withdrawal at Power Boron Dilution
a. Rod Position Alarms (Insertion Limit and Deviation)	Non-Class 1E	
b. Rod Bottom Alarm & Status Light	Non-Class 1E	
c. Rod Position Indicator	RG 1.97 Cat. 3	

REV. 1

Table 3 (continued)
Event Indicators (Page 4 of 5)

EVENT INDICATORS	SAFETY CLASSIFICATION	EVENT ASSUMED
SG Level via AFW System Level Instrumentation	Class 1E	Feedline Breaks FW System Malfunctions Locked Rotor Loss of Flow Events
Pressurizer Relief Tank		Feedline Break RCS Depressurization
a. Level Indication	RG 1.97 Cat.3	
b. Temperature Indication	RG 1.97 Cat.3	
c. Pressure Indication	RG 1.97 Cat.3	
Reactor Coolant Pump		Locked Rotor Loss of Flow Events LOCA (large)
a. Overcurrent Indication	Non-Class 1E	
b. Breaker Position	Class 1E	
c. Seal Leakoff Flow Alarm & Ind.	Non-Class 1E	
d. Vibration/Loose Parts	Non-Class 1E	
e. Seal Injection Flow Ind.	Non-Class 1E	
f. Seal ΔP Alarm & Ind.	Non-Class 1E	
g. Seal Outlet Temp.	Non-Class 1E	
Accumulator		LOCA (large) Steam Line Breaks
a. Level Indication	RG 1.97 Cat. 3	
b. Pressure Indication	RG 1.97 Cat. 3	
PZR. Safety & Relief Valve Position Indicators	RG 1.97 Cat. 2	RCS Depressurization Locked Rotor Loss of Flow Events Feed Line Breaks
PZR. Safety & Relief Valve Discharge Temperature	Non-Class 1E	RCS Depressurization Locked Rotor Loss of Flow Events Feed Line Breaks

Table 3 (continued)
Event Indicators (Page 5 of 5)

EVENT INDICATORS	SAFETY CLASSIFICATION	EVENT ASSUMED
SG Safety, Relief, & Dump Position	RG 1.97 Cat. 2	Loss of Load/Turbine Trip MSS Depressurization
Containment Pressure Monitor	RG 1.97 Cat. 1	RCS Depressurization Feed Line Breaks Steam Line Breaks LOCA (large and small)
Ice Condenser System		Feed Line Breaks Steam Line Breaks LOCA
a. Door Open Alarm	Non-Class 1E	
b. Ice Temperature Recorder	Non-Class 1E	
Containment Environment		Feed Line Breaks Steam Line Breaks LOCA
a. Temperature	RG 1.97 Cat. 1	
b. Moisture		
Feedwater System		FW System Malfunctions Feed Line Breaks
a. FW Pump Speed Monitor	Non-Class 1E	
b. FW Pump Status Indicator	Non-Class 1E	
c. FW Discharge Pressure	Non-Class 1E	
d. FW Discharge Flow	Non-Class 1E	
Reactor Bldg. Floor & Equipment Sump Level Alarm	Non-Class 1E	LOCA (large and small)

Key:

Instrument Class 1E: Seismic and environmental qualification, redundancy, and powered from a 1E power supply. 10CFR50 Appendix B QA applies.
RG 1.97 Category 1: Seismic and environmental qualification, redundancy, and powered from a 1E power supply. 10CFR50 Appendix B QA applies.
RG 1.97 Category 2: Environmental qualification. 10CFR50 Appendix B QA applies. Most, but not all, Category 2 instrumentation is powered from a 1E power source.
RG 1.97 Category 3: Non-safety related, high quality, commercial grade equipment. Some, but not all, Category 3 instrumentation is powered from a 1E power source.
Non-Class 1E: Non-safety related components. Some non-Class 1E instrumentation is powered from a 1E power source. Rev. 1

Table 4

Watts Bar Units 1 and 2
Safety Analysis Transients that use Eagle 21 Process Protection System Channels for Primary Safety Function Actuation/Diverse Automatic Backup

Transient	Primary Protection	Function	Backup Protection	Function
RWAP[1] (FSAR 15.2.2)	OTΔT	RT	HNF-PR	RT
LOL/TT[2] (FSAR 15.2.7)	Low-Low SG Level OTΔT	RT	RT on TT SG Safety Valve Steam Dump/PORV	RT Load Rejection Load Rejection
LONF[3] (FSAR 15.2.8)	Low-Low SG Level	RT/AFW	AMSAC	TT/AFW (RT on TT)
LOOP[3] (FSAR 15.2.9)	Low-Low SG Level	RT/AFW	AMSAC	TT/AFW (RT on TT)
FLB[3] (FSAR 15.4.2.2)	Low-Low SG Level	RT/AFW	AMSAC	TT/AFW (RT on TT)

Notes

1. Primary protection signal depends on the reactivity insertion rate. In general for slower reactivity insertion rates the primary reactor trip signal occurs on OTΔT, while for faster reactivity insertion rates the primary reactor trip signal is on HNF-Power Range.

2. Primary reactor trip signal depends on initial accident conditions.

3. Table only applies to operation above 50% power. Below 50% power a reactor trip does not automatically occur on a turbine trip signal. AMSAC not available below C-20 (40% RTP). See accident specific discussions in Section 2.3.

Table 5

Watts Bar Units 1 and 2
Safety Analysis Transients that use Eagle 21 Process Protection System Channels for Both Primary and Backup Safety Function Actuation

Transient	Primary Safety Signal	Function	Backup Safety Signal	Function	Non-Eagle 21 Protection, Indicators, or Alarms Available
PLOF (1 out 4 loops) (FSAR 15.2.5)	Low RC Flow	RT	----	----	RCP Protection (Table 3, Item 10) PZR Relief & Safety Valve Pos. PZR Relief & Safety Valve Temp. SG Level Monitor Core Exit Thermocouples (high)
FWM(1) (FSAR 15.2.10)	Hi-Hi SG LV	TT/ FLI	OTΔT OPΔT HNF-PR	RT RT RT	SG Level Monitor FW System Monitors (Table 3, Item 18)
RCS Depr. (FSAR 15.2.12)	PZR PRESS (low)	RT/SI	OTΔT	RT	Containment Pressure Monitor PZR Relief & Safety Valve Pos. PZR Relief & Safety Discharge Temp. PZR Relief Tank (Table 3, Item 9)
MSS Depr.(2) (FSAR 15.2.13)	PZR PRESS (low)	RT/SI	OPΔT Low SL PRESS Hi-Hi CONT PRESS	RT ESF SLI	HNF and HNF Rate - RT SG Level Monitor SG Valve or Steam Dump Pos. Ind. Core Exit Thermocouples (low)
LOF-LR (FSAR 15.4.4)	Low RC Flow	RT	OTΔT PZR PRES (hi)	RT RT	RCP Protection (Table 3, Item 10) PZR Relief & Safety Valve Pos. PZR Releif & Safety Valve Temp. SG Level Monitor Core Exit Thermocouples (High)
SLB-CR(2) (FSAR 15.4.2.1)	Low SL PRESS	ESF	Hi-Hi CONT PRESS PZR PRESS (lo) OPΔT	SLI SI RT	SG Level Monitor HNF & HNF Rate (Pos.) Accumulator Level & Press. Ind. Core Exit Thermocouples (Low) Containment Pressure Monitor
SLB-CONT. (FSAR 6.2.1.3.10)	Low SL PRESS OPΔT PZR PRESS (low)	ESF RT	Hi-Hi CONT PRESS Hi CONT PRESS	SLI SI	SG Level Monitor HNF & HNF Rate (Pos.) Ice Condenser System (Table 3, Item 16) Core Exit Thermocouples (Low) Accumulator Level & Press. Ind. Containment Pressure Monitor
SGTR (FSAR 15.4.3)	OTΔT	RT	PZR PRESS (low)	RT/SI	SL Radiation Monitor Condenser Vacuum Radiation Monitor SG Blowdown Radiation Monitor
LOCA(3) (FSAR 15.3.1) (FSAR 15.4.1)	PZR PRESS (low)	ESF/ RT	Hi CONT. PRES RT on ESF	ESF RT	Containment Radiation Monitors Reactor Bld. Floor & Equip. Sump Alarm Core Exit Thermocouples (high) Containment Pressure Monitor Containment Environment Monitors RCS Flange Leakoff Temperature Ice Condenser System (Table 3, Item 16)

Table 5 (Continued)

Watts Bar Units 1 and 2
Safety Analysis Transients that use Eagle 21 Process Protection System Channels for Both Primary and Backup Safety Function Actuation

Notes

1. Feedwater isolation is required to prevent excessive moisture carryover to the turbine and water in the steam pipes (which could cause a steam line break event). For a excessive flow feedwater malfunction event automatic actuation of feedwater isolation is not available outside EAGLE 21. Indications are available to the operator to alert this condition for manual control.

2. Steam line break cases analyzed at-power, without Eagle 21 Process Protection System functions, would receive high neutron flux reactor trip signals (Nuclear Instrumentation System).

3. Large Break LOCA analysis assumes that the rods do not trip.

Table 6

Table of Acronyms and Abbreviations

AFW	- Auxiliary Feedwater
AMSAC	- ATWS Mitigation System Actuation Circuitry
ATWS	- Anticipated Transient Without Scram
CONT	- Containment
CR	- Core Response
Depr.	- Depressurization
DNBR	- Departure From Nucleate Boiling Ratio
ESF	- Engineered Safety Feature
FLB	- Feedline Break
FLI	- Feedline Isolation
FSAR	- Final Safety Analysis Report
FW	- Feedwater
FWM	- Feedwater Malfunction
HNF	- High Neutron Flux
LOCA	- Loss of Coolant Accident
LOF	- Loss of Flow
LOL	- Loss of Load
LONF	- Loss of Normal Feedwater
LOOP	- Loss of Offsite Power
LR	- Locked Rotor
LV	- Level
MSS	- Main Steam System
NIS	- Nuclear Instrumentation System
NSAC	- Nuclear Safety Analysis Center
OTΔT	- Overtemperature ΔT
OPΔT	- Overpower ΔT
PLOF	- Partial Loss of Flow
PORV	- Power Operated Relief Valve
Pos.	- Position
PRA	- Probalistic Risk Assessment
PRESS	- Pressure
PSRV	- Pressurizer Safety Relief Valve
PZR	- Pressurizer
RC	- Reactor Coolant
RCCA	- Rod Cluster Control Assembly
RCP	- Reactor Coolant Pump
RCS	- Reactor Coolant System
RT	- Reactor Trip
RTP	- Rated Thermal Power
RWST	- Refueling Water Storage Tank
SG	- Steam Generator
SGL	- Steam Generator Level
SGTR	- Steam Generator Tube Rupture
SL	- Steam Line
SLB	- Steam Line Break
SI	- Safety Injection
SLI	- Steam Line Isolation
TT	- Turbine Trip
TTD	- Trip Time Delay