7.1 INTRODUCTION

7.1.1 IDENTIFICATION OF SAFETY RELATED SYSTEMS

This Subsection is divided into two subsections: 7.1.1a, listing systems provided with the nuclear steam supply system (NSSS), and 7.1.1b, listing systems provided by others (non-NSSS). Division of responsibility is indicated in this manner throughout Chapter 7 where required.

Systems added by PP&L may be listed in the NSSS Subsection if the PP&L supplied systems are modifications of or additions to one or more NSS system, or if the PP&L supplied systems serve a specific redundancy or diversity function with respect to one or more NSS system.

This arrangement is used to preserve the standard format of the NSSS supplier. Subsection assignments in the NSSS format which were assigned to non-NSSS will be indicated with the sentence, "This Subsection was not used."

7.1.1a IDENTIFICATION OF SAFETY-RELATED SYSTEMS PROVIDED WITH THE NSSS

7.1.1a.1 General (NSSS)

Instrumentation and control systems supplied by General Electric are designated as either power generation systems or safety systems, depending on their function. Some portions of a system may have a safety function while other portions of the same system may be classified as power generation. A description of this system of classification can be found in Appendix 15A.

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The systems presented in Chapter 7.0 are also classified according to Regulatory Guide 1.70, Revision 2; namely, Reactor Protection (Trip) System, Engineered Safety Feature Systems, Safe Shutdown Systems, Safety-Related Display Instrumentation, Other Systems Required for Safety, and Control Systems Not Required for Safety. Table 7.1-1 lists the systems under each of these classifications and identifies the designer and/or the supplier. *Table 7.1-2 identifies instrumentation and control systems that are similar to those of a nuclear power plant of similar design that has recently received NRC design or operation approval through the issuance of either a construction permit or an operating license. Differences and their effect on safety-related systems are also identified in Table 7.1-2.*

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7.1.1a.2 Identification of Individual Systems

The following are safety-related systems:

(1) Reactor Protection System (RPS)
(2) Primary Containment and Reactor Vessel Isolation Control System (PCRVICS)
(3) Emergency Core Cooling Systems (ECCS)
(4) Neutron Monitoring System (NMS)
(5) Process Radiation Monitoring System
(6) Reactor Core Isolation Cooling (RCIC) System
(7) Standby Liquid Control System (SLCS)
(8) NSSS Leak Detection Systems
(9) RHRS Reactor Shutdown Mode (RHR)
(10) Safety-Related Display
(11) RHRS - Containment Spray Cooling Mode
(12) Recirculation Pump Trip (RPT) System
(13) RHRS Suppression Pool Cooling Mode
(14) ATWS Mitigation Capability - Alternate Rod Injection (ARI) System – Non-NSSS

7.1.1a.3 Classification (NSSS)

7.1.1a.3.1 Safety-Related Systems (NSSS)

Safety-related systems provide actions necessary to assure safe shutdown, to protect the integrity of radioactive material barriers, and/or to prevent the release of radioactive material in excess of allowable dose limits. These systems may be components, groups of components, systems, or groups of systems. Engineered Safety Feature (ESF) systems are included in this category. ESF systems have a sole function of mitigating the consequences of design basis accidents.

The Isolated Condenser Treatment Method (ICTM) is an ESF that mitigates the consequences of post LOCA, MSIV leakage. The ICTM is an approved Non-Safety-Related alternative to Regulatory Guide 1.96 for treating MSIV leakage. Therefore, the ICTM is an ESF that is not included in the Safety-Related category.

7.1.1a.3.2 Power Generation (Non-Safety) Systems (NSSS)

Power generation systems are not required to protect the integrity of radioactive material barriers and/or prevent the release of radioactive material beyond allowable dose limits. The instrumentation and control portions of these systems may, by their actions, prevent the plant from exceeding preset limits which would otherwise initiate action of the safety-related systems.

7.1.1a.3.3 Design Basis (NSSS)

The various NSS Systems may have both a safety design basis and a power generation design basis depending on their function. The safety design basis states in functional terms the unique design requirements that establish limits for the operation of the system. The general functional requirements portion of the safety design basis presents those requirements which have been determined to be sufficient to ensure the adequacy and reliability of the system from a safety viewpoint. These requirements have been introduced into various codes, criteria, and regulatory requirements. Safety and Power generation design bases are discussed in Subsection 15A.2.2.

7.1.1a.3.4 Specific Regulatory Requirements (NSSS)

The NSS Systems have been examined with respect to specific regulatory requirements which are applicable to the subject instrumentation and controls systems. These regulatory requirements include:

(a) Industry codes and standards
(b) Title 10 Code of Federal Regulations
(c) NRC electrical instrumentation and control systems branch technical positions, and
(d) NRC Regulatory Guides

The specific regulatory requirements applicable to NSSS's instrumentation and control are specified in Table 7.1-3. The RPS, PCRVICS, ECCS, and Leak Detection Systems have been reduced to the subsystem level, and the applicable regulatory requirements for each have been specified. This information is contained in Sections 7.2, 7.3, and 7.6.

7.1.1b IDENTIFICATION OF SAFETY RELATED SYSTEMS NOT PROVIDED WITH THE NSSS (NON-NSSS)

7.1.1b.1 Engineered Safety Feature Systems (Non-NSSS)

Instrumentation and controls for Engineered Safety Feature (ESF) Systems are identified as follows:

(1) Primary containment isolation controls
(2) Combustible gas control system
(3) Primary containment vacuum relief
(4) Standby gas treatment system
(5) Reactor building recirculation system
(6) Reactor building isolation and HVAC support

(7) Habitability systems including control room envelope isolation and supporting HVAC systems

  a) Control room HVAC
  b) Control structure HVAC
  c) Computer room cooling system
  d) Emergency outside air supply
  e) Battery room exhaust system

7.1.1b.2 Engineered Safety Feature Auxiliary Support Systems (Non-NSSS)

Instrumentation and controls for ESF auxiliary support systems are identified as follows:

(1) Emergency service water (ESW)
(2) RHR service water (RHRSW)
(3) Containment instrument gas
(4) Standby power
(5) Heating, ventilating, and air conditioning for ESC areas:

  a) Standby gas treatment equipment room
  b) Diesel generator buildings
  c) SEW pumphouse
  d) ESC switchgear room
  e) ECCS unit coolers
  f) Drywell unit coolers
  g) Control structure chilled water system

7.1.1b.3 Systems Required for Safe Shutdown (Non-NSSS)

Instrumentation an controls for the system required for safe shutdown are identified as:

(1) Reactor shutdown outside the main control room

7.1.1b.4 Safety Related Display Instrumentation (Non-NSSS)

Safety-related display instrumentation described provides operator information for normal plant operations and to perform manual safety functions.

Safety-related displays are provided for monitoring the following:

(1) ESF systems
(2) ESF auxiliary support systems
(3) Plant processes
(4) Bypass and inoperable status
(5) Post-accident monitoring
(6) Reactor shutdown outside the control room

7.1.1b.5 Other Systems Required for Safety (Non-NSSS)

Instrumentation and controls for all other systems required for safety are identified as:

(1) Drywell entry purge
(2) Containment atmosphere monitoring
(3) NSSS to non-NSSS interlocks for standby power start (diesels)
(4) Process and effluent radiation monitoring

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7.1.1b.6 Non-NSSS Systems Similar to Plants Which Have An Operating License

Non-NSSS ESF Systems and auxiliary support systems, while functionally similar to other plants, are not sufficiently similar to other plants to allow meaningful comparison. However, certain equipment, noted below and in Table 7.1-2, was furnished as part of the NSSS Contract and therefore is similar to the equipment furnished in the plants noted in the table. The application of the equipment should be considered unique to Susquehanna SES. Similar equipment:

Process Radiation Monitoring Equipment
Area Radiation Monitoring Equipment
Primary Containment and Reactor Vessel Isolation
Control System (See Note 2 on Table 7.1-2)

END HISTORICAL

7.1.2 IDENTIFICATION OF SAFETY CRITERIA

7.1.2a.1 Identification of Safety Criteria for NSSS, General

Design bases and criteria for instrumentation and control equipment design are based on the need to have the system perform its intended function while meeting the requirements of applicable general design criteria, regulatory guides, industry standards, and other documents.

7.1.2a.1.1 Reactor Protection System (RPS) – Instrumentation and Control

7.1.2a.1.1.1 Safety Design Bases

7.1.2a.1.1.1.1 General Functional Requirements

The reactor protection system (RPS) is designed to meet the following functional requirements:

(1) The RPS shall initiate a reactor scram with precision and reliability to prevent or limit fuel damage following abnormal operational transients.

(2) The RPS shall initiate a scram with precision and reliability to prevent damage to the reactor coolant pressure boundary (RCPB) as a result of excessive internal pressure; that is, to prevent reactor vessel pressure from exceeding the limit allowed by applicable industry codes.

(3) To limit the uncontrolled release of radioactive materials from the fuel assembly or RCPB, the RPS shall precisely and reliably initiate a reactor scram on gross failure of either of these barriers.

(4) To detect conditions that threaten the fuel assembly or RCPB, the RPS inputs shall be derived from variables that are true, direct measures of operational conditions.

(5) The RPS shall respond correctly to the sensed variables over the expected range of magnitudes and rates of change.

(6) A sufficient number of sensors shall be provided for monitoring essential variables that have spatial dependence.

(7) The following bases assure that the RPS is designed with sufficient reliability:

a. If failure of a control or regulating system causes a plant condition that requires a reactor scram but also prevents action by necessary reactor protection system channels, the remaining portions of the RPS shall meet the requirements 1, 2, and 3 above.

b. Loss of one power supply shall neither cause nor prevent a reactor scram.

c. Once initiated, a RPS action shall go to completion. Return to normal operation shall require deliberate operator action.

d. There shall be sufficient electrical and physical separation between redundant instrumentation and control equipment monitoring the same variable to prevent environmental factors, electrical transients, or physical events from impairing the ability of the system to respond correctly.

e. Safe Shutdown Earthquake, as amplified by building and supporting structures, shall not impair the ability of the RPS to initiate a reactor scram.

f. No single failure within the RPS shall prevent proper reactor protection system action, when required, to satisfy safety design bases 1, 2, and 3 above.

g. Any one intentional bypass, maintenance operation, calibration operation, or test to verify operational availability shall not impair the ability of the RPS to respond correctly.

h. The system shall be designed to that the required number of sensors for any monitored variable exceeding the scram setpoint will initiate an automatic scram.

(8) The following bases reduce the probability that RPS operational reliability and precision will be degraded by operator error:

a. Access to trip settings, component calibration controls, test points, and other terminal points shall be under the control of plant operations supervisory personnel.

b. Manual bypass of instrumentation and control equipment components shall be under the control of the control room operator. If the ability to trip some essential part of the system has been bypassed, this fact shall be continuously annunciated in the main control room.

7.1.2a.1.1.1.2 Specific Regulatory Requirements

The specific regulatory requirements applicable to this system are shown in Tables 7.1-3 and 7.1-4. The degree of conformance to these requirements is discussed in the analysis section for this system.

7.1.2a.1.1.2 Power Generation (Non-Safety) Design Bases

The setpoints, power sources, and control and instrumentation are arranged in such a manner as to preclude spurious scrams.

7.1.2a.1.2 Primary Containment and Reactor Vessel Isolation Control System (PCRVICS) - Instrumentation and Controls

7.1.2a.1.2.1 Safety Design Bases

7.1.2a.1.2.1.1 General Functional Requirements

The following functional design bases have been implemented in the primary containment and reactor vessel isolation control system:

(1) To limit the release of radioactive materials to the environs, the PCRVICS shall, with precision and reliability, initiate timely isolation of penetrations through the primary containment whenever the values of monitored variables exceed preselected operational limits.

(2) To provide assurance that important variables are monitored with a precision sufficient to fulfill Safety Design Basis (1), the PCRVICS shall respond correctly to the sensed variables over the expected design range of magnitudes and rates of change.

(3) To provide assurance that important variables are monitored to fulfill Safety Design Basis (1), a sufficient number of sensors shall be provided for monitoring essential variables.

(4) To provide assurance that conditions indicative of a failure of the RCPB are detected to fulfill Safety Design Basis (1), PCRVICS inputs shall be derived from variables that are true, direct measures of operational conditions.

(5) The time required to close the main steamline isolation valves shall be short to minimize the loss of coolant from a steamline break outside containment.

(6) The time required to close the main steamline isolation valves shall not be so short that inadvertent isolation of steamlines causes a transient more severe than that resulting from closure of the turbine stop valves coincident with failure of the turbine bypass system. This ensures that the main steam isolation valve closure speed is compatible with the ability of the RPS to protect the fuel assembly and RCPB.

(7) To provide assurance that the closure of automatic isolation valves is initiated when required to fulfill Safety Design Basis (1), the following safety design bases are specified for the systems controlling automatic isolation valves:

   a. Any one failure, maintenance operation, calibration operation, or test to verify operational availability shall not impair the functional ability of the isolation control system.

   b. The system shall be designed so that the required number of sensors for any monitored variable exceeding the isolation setpoint will initiate automatic isolation.

   c. Where a plant condition that requires isolation can be brought on by a failure or malfunction of a control or regulating system, and the same failure or malfunction prevents action by one or more isolation control system channels designed to provide protection against the unsafe condition, the remaining portions of the isolation control system shall meet the requirements of Safety Design Bases (1), (2), (3), and (7)a.

   d. The power supplies for the PCRVICS shall be arranged so that loss of one supply cannot prevent automatic isolation when required.

   e. The system shall be designed so that, once initiated, automatic isolation action goes to completion. Return to normal operation after isolation action shall require deliberate operator action.

f. There shall be sufficient electrical and physical separation of wiring and piping between trip channels monitoring the same essential variable to prevent environmental factors, electrical faults, and physical events from impairing the ability of the system to respond correctly.

g. SSE ground motions shall not impair the ability of the PCRVICS to initiate automatic isolation.

(8) The following safety design basis is specified to assure that the isolation of main steamlines is accomplished:

a. The isolation valves in each of the main steamlines shall not rely on electrical power to achieve closure.

(9) To reduce the probability that the operational reliability of the PCRVICS will be degraded by operator error, the following safety design bases are specified for automatic isolation valves.

a. Access to all trip settings, component calibration controls, test points, and other terminal points for equipment associated with essential monitored variables shall be under the physical control or supervision of plant operations supervisory personnel.

b. The means for bypassing trip channels, trip logics, or system components shall be under the control of plant operations supervisory personnel. If the ability to trip some essential part of the system has been bypassed, this fact shall be continuously indicated in the control room.

(10) To provide the operator with a means to take action that is independent of the automatic isolation functions in the event of a failure of the RCPB, it shall be possible for the operator to manually initiate isolation of the primary containment and reactor vessel from the main control room.

(11) The following bases are specified to provide the operator with the means to assess the condition of the PCRVICS and to identify conditions indicative of a gross failure of RCPB.

a. The PCRVICS shall be designed to provide the operator with information pertinent to the status of the system.

b. Means shall be provided for prompt identification of trip channel and trip system responses.

(12) It shall be possible to check the operational availability of each trip channel and trip logic during reactor operation.

7.1.2a.1.2.2 Specific Regulatory Requirements

The specific regulatory requirements applicable to this system are shown in Tables 7.1-3 and 7.1-5. The degree of conformance to these requirements is discussed in the analysis section for this system.

7.1.2a.1.3 Emergency Core Cooling Systems – Instrumentation and Controls

7.1.2a.1.3.1 Safety Design Bases

7.1.2a.1.3.1.1 General Functional Requirements

The emergency core cooling systems (ECCS) control and instrumentation shall be designed to meet the following functional safety design bases:

(1) Automatically initiate and control the ECCS to prevent excessive fuel cladding temperatures.

(2) Respond to a need for emergency core cooling, regardless of the physical location of the malfunction or break that causes the need.

(3) The following safety design bases are specified to limit dependence on operator judgement in times of stress:

   a. With one exception, the ECCS shall respond automatically so that no action is required of plant operators within 20 minutes after a LOCA. The only operator action assumed in the Section 6.3 ECCS analysis is that a RHR heat exchanger is placed in service within 20 minutes into the accident.

   b. The performance of the ECCS shall be indicated by main control room instrumentation.

   c. Facilities for manual control of the ECCS shall be provided in the main control room.

7.1.2a.1.3.1.2 Specific Regulatory Requirements

The specific regulatory requirements applicable to this system are shown in Tables 7.1-3, 7.1-6 and 7.1-7. The degree of conformance to these requirements is discussed in the analysis section for this system.

7.1.2a.1.4 Neutron Monitoring System - Instrumentation and Controls

7.1.2a.1.4.1 Source Range Monitor (SRM) Subsystem

This control system is not required for safety.

7.1.2a.1.4.1.1 Power Generation (Non-Safety) Design Bases

The source range monitor (SRM) subsystem meets the following power generation design bases:

(1) Neutron sources and neutron detectors together shall result in a signal-to-noise ratio of at least 2:1 and a count rate of at least three counts per second with all control rods fully inserted prior to initial power operation.

(2) The SRM shall be able to perform the following functions:

   a. Indicate a measurable increase in output signal from at least one detecting channel before the reactor period is less than 20 seconds during the worst possible startup rod withdrawal conditions.

   b. Indicate substantial increases in output signals with the maximum permitted number of SRM channels out of service during normal reactor startup operations.

   c. The SRM channels shall be on scale when the IRM first indicates neutron flux during a reactor startup.

   d. Provide a measure of the time rate of change of the neutron flux (reactor period) for operational convenience.

   e. Generate interlock signals to block control rod withdrawal if the count rate exceeds a preset value or falls below a preset limit (if the IRMs are not above the second range) or if certain electronic failures occur.

(3) Perform its function in the maximum normal thermal and radiation environment.

(4) Loss of a single power bus will not disable the monitoring and alarming functions of all the available monitors.

7.1.2a.1.4.2 Intermediate Range Monitor (IRM) Subsystem

7.1.2a.1.4.2.1 Safety Design Basis

The IRM generates a trip signal that can be used to prevent fuel damage resulting from anticipated or abnormal operational transients that occur while operating in the intermediate power range. The independence and redundancy incorporated in the design of the IRM is consistent with the safety design bases of the RPS.

7.1.2a.1.4.2.2 Specific Regulatory Requirements

The specific regulatory requirements applicable to this system are shown in Table 7.1-3. The degree of conformance to these requirements is discussed in the analysis section for this system.

7.1.2a.1.4.2.3 Power Generation (Non-Safety) Design Bases

The IRM generates an interlock signal to block rod withdrawal if the IRM reading exceeds a preset value or if the IRM is not operating properly. The IRM is designed so that overlapping neutron flux indications exist with the SRM and APRM subsystems.

7.1.2a.1.4.3 Local Power Range Monitor (LPRM) Subsystem

7.1.2a.1.4.3.1 Safety Design Basis

The LPRMs provide input to the APRM Subsystem (see Subsection 7.1.2a.1.4.4) and the OPRM Subsystem (see Subsection 7.1.2a.1.4.7).

7.1.2a.1.4.3.2 Power Generation (Non-Safety) Design Bases

The LPRM supplies:

(1) Signals to the APRM that are proportional to the local neutron flux at various locations within the reactor core;

(2) Signals to alarm high or low local neutron flux;

(3) Signals proportional to the local neutron flux to drive indicating meters and auxiliary devices to be used for operator evaluation of power distribution, local heat flux, minimum critical power ratio and fuel burnup rate;

(4) Signals to the RBM to indicate changes in local relative neutron flux during the movement of control rods.

7.1.2a.1.4.4 Average Power Range Monitor (APRM) Subsystem

7.1.2a.1.4.4.1 Safety Design Basis

Under the worst permitted input LPRM bypass conditions, the APRM is capable of generating a trip signal in response to average neutron flux increases (or thermal-hydraulic instability caused by Power Oscillations) in time to prevent fuel damage. The independence and redundancy incorporated into the design of the APRM is consistent with the safety design bases of the reactor protection system.

The APRM provides power level and drive flow signals to the OPRM.

7.1.2a.1.4.4.2 Specific Regulatory Requirements

The specific regulatory requirements applicable to this system are shown in Table 7.1-3. The degree of conformance to these requirements is discussed in the analysis section for this system.

7.1.2a.1.4.4.3 Power Generation (Non-Safety) Design Basis

The APRM provides the following functions:

(1) A continuous indication of average reactor power (neutron flux) from a few percent to 125% of rated reactor power;

(2) Interlock signals for blocking further rod withdrawal to avoid an unnecessary scram actuation;

(3) A reference power level for controlling reactor recirculation system flow;

(4) A reactor thermal power signal derived from each APRM channel which approximates the dynamic effects of the fuel;

(5) A reference power level for the Rod Block Monitor subsystem.

7.1.2a.1.4.5 Traversing Incore Probe (TIP) Subsystem

This control system is not required for safety. See Subsections 7.7.1.6 and 7.7.2.6.

7.1.2a.1.4.5.1 Power Generation Design Bases

The TIP subsystem meets the following power generation design bases:

(1) Provide a signal proportional to the axial neutron flux distribution at selected axial intervals over the regions of the core where LPRM detector assemblies are located. This signal shall be of high precision to allow reliable calibration of LPRM gains.

(2) Provide accurate indication of the position of the flux measurement to allow pointwise or continuous measurement of the axial neutron flux distribution.

7.1.2a.1.4.6 Rod Block Monitor (RBM) Subsystem

This subsystem is not required for safety. See Subsections 7.7.1.11 and 7.7.2.11.

7.1.2a.1.4.6.1 Power Generation (Non-Safety) Design Bases

The rod block monitor (RBM) subsystem meets the following power generation design bases:

(1) Prevent local fuel damage that may result from a single rod withdrawal error.

(2) Provide a signal used by the operator to evaluate the change in the local relative power level during control rod movement.

7.1.2a.1.4.7 Oscillation Power Range Monitor (OPRM) Subsystem

7.1.2a.1.4.7.1 Safety Design Basis

Under the limiting input LPRM bypass conditions, the OPRM is capable of generating a trip signal in response to the thermal-hydraulic induced neutron flux instabilities in time to prevent fuel damage. The independence and redundancy incorporated into the design of the OPRM is consistent with the safety design of the reactor protection system.

The OPRM RPS trip function is enabled when the reactor is in a low flow-high power configuration. Exact limits for this region are specified in the Core Operating Limits Reports (COLR).

7.1.2a.1.4.7.2 Specific Regulatory Requirements

The specific regulatory requirements applicable to this system are shown in Table 7.1-3. The degree of conformance to these requirements is discussed in the analysis section for this system.

7.1.2a.1.5 Refueling Interlocks - Instrumentation and Controls

Refueling interlocks are not required for safety. See Subsections 7.7.1.10 and 7.7.2.10.

7.1.2a.1.6 Reactor Manual Control System – Instrumentation and Controls

This system is not required for safety. See Subsections 7.7.1.2 and 7.7.2.2.

7.1.2a.1.6.1 Specific Regulatory Requirements

The specific regulatory requirements applicable to this system are shown in Table 7.1-3. The degree of conformance to these requirements is discussed in the analysis section for this system.

The OPRM provides signals to the plant computer indicating process and conditions and OPRM data for operator evaluation.

7.1.2a.1.6.2 Power Generation (Non-Safety) Design Bases

The reactor manual control system is designed to meet the following power generation design bases:

(1) Inhibit control rod withdrawal following erroneous control rod manipulations so that reactor protection system action (scram) is not required.

(2) Inhibit control rod withdrawal in time to prevent local fuel damage as a result of erroneous control rod manipulation.

(3) Inhibit control rod movement whenever such movement would result in operationally undesirable core reactivity conditions or whenever instrumentation is incapable of monitoring the core response to rod movement.

(4) Limit the potential for inadvertent rod withdrawals leading to reactor protection system action by designing the reactor manual control system in such a way that deliberate operator action is required to effect a continuous rod withdrawal.

(5) Provide the operator with the means to achieve prescribed control rod patterns, and provide information pertinent to the position and motion of the control rods in the main control room.

7.1.2a.1.7 Through 7.1.2a.1.10

These subsections were not used.

7.1.2a.1.11 Process Radiation Monitoring System - Instrumentation and Controls

7.1.2a.1.11.1 Main Steamline Radiation Monitoring Subsystem

7.1.2a.1.11.1.1 Safety Design Basis

The main steamline radiation monitoring subsystem is designed to meet the following safety design bases:

(1) The subsystem is able to detect a gross release of fission products from the fuel under any anticipated operating combination of main steamlines.

(2) The subsystem shall indicate a gross release of fission products from the fuel.

(3) On detection of a gross release of fission products from the fuel:

   a. The subsystem shall initiate a main control room annunciator (alarm).

   b. The subsystem shall provide the trip signal to the PCRVICS to isolate the containment via the reactor coolant sample valves. The subsystem shall trip the Mechanical Vacuum Pump (MVP) and isolate its suction valves.

7.1.2a.1.11.1.2 Specific Regulatory Requirements

The specific regulatory requirements applicable to this system are shown in Tables 7.1-3 and 7.1-8. The degree of conformance to these requirements is discussed in the analysis section for this system.

7.1.2a.1.11.1.3 Power Generation (Non-Safety) Design Basis

The main steamline radiation monitoring subsystem is designed to display in the main control room an indication of gross gamma radiation level at the main steam tunnel.

7.1.2a.1.12 through 7.1.2a.1.17

These subsections were not used.

7.1.2a.1.18 Reactor Core Isolation Cooling (RCIC) System – Instrumentation and Controls

7.1.2a.1.18.1 Safety Design Bases

(1) The system is capable of maintaining sufficient coolant in the reactor vessel in case of an isolation with a loss of main feedwater flow.

(2) Provisions are made for automatic and remote manual operation of the system.

(3) Components of the RCIC system are designed to satisfy Seismic Category I design requirements.

(4) To provide a high degree of assurance that the system shall operate when necessary, the power supply for the system is from immediately available energy sources of high reliability.

(5) To provide a high degree of assurance that the system shall operate when necessary, provision is made so that periodic testing can be performed during plant operation.

7.1.2a.1.18.2 Specific Regulatory Requirements

RCIC is considered a Safe Shutdown System rather than an ECCS. The specific regulatory requirements applicable to this system are shown in Table 7.1-3. The degree of conformance to these requirements is discussed in the analysis section for this system.

7.1.2a.1.19 Standby Liquid Control System (SLCS) - Instrumentation and Controls

7.1.2a.1.19.1 Safety Design Basis

This system is capable of shutting the reactor down from full power to cold shutdown and maintaining the reactor in a subcritical state at atmospheric temperature and pressure conditions by pumping sodium pentaborate, a neutron absorber, into the reactor.

The system will also be used to buffer suppression pool pH to prevent Iodine re-evolution following a postulated design basis loss of coolant accident.

7.1.2a.1.19.2 Specific Regulatory Requirements

The specific regulatory requirements applicable to this system are shown in Table 7.1-3. The degree of conformance to these requirements is discussed in the analysis section for this system.

7.1.2a.1.19.3 Power Generation (Non-Safety) Design Basis

There are no power generation bases for this system.

7.1.2a.1.20 through 7.1.2a.1.23

These subsections are not used.

7.1.2a.1.24 NSSS Leak Detection System – Instrumentation and Controls

7.1.2a.1.24.1 Reactor Coolant Pressure Boundary Leakage Detection

7.1.2a.1.24.1.1 Safety Design Bases

The safety design bases for the leak detection systems are as follows:

(1) Signals are provided to permit isolation of abnormal leakage before the results of this leakage become unacceptable.

(2) The unacceptable results are as follows:

   a. A threat of significant compromise to the reactor coolant pressure boundary.

   b. A leakage rate in excess of the coolant makeup capability to the reactor vessel.

7.1.2a.1.24.1.2 Specific Regulatory Requirements

The specific regulatory requirements applicable to this system are shown in Tables 7.1-3 and 7.1-9. The degree of conformance to these requirements is discussed in the analysis section for this system.

7.1.2a.1.24.1.3 Power Generation (Non-Safety) Design Basis

A means is provided to detect abnormal leakage from the RCPB.

7.1.2a.1.25 RHRS - Reactor Shutdown Cooling Subsystem – Instrumentation and Controls

7.1.2a.1.25.1 Safety Design Bases

The reactor shutdown cooling mode function of the RHR system is designed to meet the following functional design bases:

(1) Instrumentation and controls are provided that will enable the system to remove the residual heat (decay heat and sensible heat) from the reactor vessel during normal shutdown.

(2) Manual controls of the shutdown cooling system are provided in the main control room and the remote shutdown panel.

(3) Performance of the shutdown cooling system is indicated by main control room instrumentation and similar instrumentation in the remote shutdown panel.

7.1.2a.1.25.2 Specific Regulatory Requirements

The specific regulatory requirements applicable to this system are shown in Table 7.1-3. The degree of conformance to these requirements is discussed in the analysis section for this system.

7.1.2a.1.25.3 Power Generation (Non-Safety) Design Bases

The reactor shutdown cooling mode of the residual heat removal system (RHR) shall meet the following power generation design bases:

(1) Provide cooling for the reactor during the shutdown operation when the vessel pressure is below approximately 100 psig.

(2) Cool the reactor water to a temperature which is practical for refueling and servicing operation.

(3) Provide means for reactor head cooling by diverting part of the shutdown flow to a nozzle in the vessel head. This flow will condense the steam generated from the hot walls of the vessel while it is being flooded, thereby keeping system pressure down.

7.1.2a.1.26 through 7.1.2a.1.29

These subsections are not used.

7.1.2a.1.30 ATWS Mitigation Capability – Instrumentation and Controls

7.1.2a.1.30.1 Special Event Design Basis

The ability of the plant to accommodate anticipated transient without scram is defined and examined in Section 15.8 and Appendix 15A (Event 51).

Mitigation of an ATWS event resulting from an electrical or electromechanical failure of the Reactor Protection System is accomplished by the ATWS-Recirculation Pump Trip System (ATWS-RPT) and by the Alternate Rod Injection (ARI) System. These systems are capable of reducing reactivity by (1) tripping the recirculation pumps via the ATWS-RPT system, and (2) providing an alternate means (ARI) to rapidly insert the control rods to achieve and maintain a subcritical configuration.

The specific design basis requirements for these systems are given in License Topical Report NEDE-31096-A and in Subsection 7.2.3.

7.1.2a.1.30.2 Specific Regulatory Requirements

Regulatory requirements for ATWS-RPT and ARI systems are given in 10CFR50.62. The degree of conformance to these requirements is discussed in the analysis sections for these systems.

7.1.2a.1.31 Safety-Related Display - Instrumentation

7.1.2a.1.31.1 Safety Design Basis

The necessary display instrumentation shall be available to reactor operator in the main control room to determine and accomplish all the required manual control actions consistent with safe plant operation.

7.1.2a.1.31.2 Specific Regulatory Requirements

The specific regulatory requirements applicable to this system are shown in Table 7.1-3. The degree of conformance to these requirements is discussed in the analysis section for this system.

7.1.2a.1.31.3 Power Generation (Non-Safety) Design Basis

Sufficient and reliable display instrumentation shall be provided such that all the expected power operation actions and maneuvers can be reasonably accomplished by the reactor operator from the main control room.

7.1.2a.1.32 through 7.1.2a.1.34

These sections were not used.

7.1.2a.1.35 RHRS - Containment Spray Cooling System - Instrumentation and Controls

7.1.2a.1.35.1 Safety Design Basis

The containment spray cooling mode function of the RHR system is designed to meet the following functional safety design bases:

a. Instrumentation and controls are provided that will sense containment and drywell pressures and enable the system to provide condensation of steam in the containment air volume during a transient or accident event.

b. All manual controls of the containment spray subsystem are provided in the control room.

c. Performance of the containment spray subsystem is indicated by control room instrumentation.

7.1.2a.1.35.2 Specific Regulatory Requirements

The specific regulatory requirements applicable to this system are shown in Table 7.1-3. The degree of conformance to these requirements is discussed in the analysis section for this system.

7.1.2a.1.35.3 Power Generation Design Basis

There are no Power Generation Design Basis for the RHR Containment Spray Cooling System.

7.1.2a.1.36 This subsection is not used.

7.1.2a.1.37 Recirculation Pump Trip (RPT) - Instrumentation and Controls

7.1.2a.1.37.1 Safety Design Bases

The Recirculation Pump Trip is designed to meet the following safety design bases:

(1) Instrumentation and controls are provided that will cause both recirculation pumps to trip when the main turbine trips or a generator load rejection occurs. The RPT will occur automatically in order to ensure that the reactor core remains within the conservative thermal hydraulic limits during certain abnormal operational transients.

(2) Operational performance is indicated by main control room instrumentation.

7.1.2a.1.37.2 Specific Regulatory Requirements

The specific regulatory requirements applicable to this system are shown in Table 7.1-3. The degree of conformance to these requirements is discussed in the analysis section for this system.

7.1.2a.1.37.3 Power Generation (Non-Safety) Bases

There are no power generation bases for this system.

7.1.2a.1.38 RHRS - Suppression Pool Cooling System - Instrumentation and Controls

7.1.2a.1.38.1 Safety Design Bases

Instrumentation and controls are provided to allow the reactor operator to manually initiate suppression pool cooling to ensure that the pool temperature immediately after any relief valve discharge to the pool does not exceed the pre-established pool temperature limit.

7.1.2a.1.38.2 Specific Regulatory Requirements

The specific regulatory requirements applicable to this system are shown in Table 7.1-3. The degree of conformance to these requirements is discussed in the analysis section for this system.

7.1.2a.1.38.3 Power Generation (Non-Safety) Design Basis

There are no power generation design bases for this system.

7.1.2b.1 Identification of Safety Criteria for non-NSSS, General

Design bases for all the safety related instrumentation of the systems are presented in the section of this chapter that discusses the system to which the bases apply.

Certain Non-NSSS ESF systems conform to the following:

(1) ESF systems generally conform to IEEE Standard 279-1971. Detailed discussion of extent of conformance is in Subsections 7.3.1b and 7.3.2b.

(2) The Operational Quality Assurance program is discussed in Section 17.2.

(3) General design criteria for nuclear power plants, Appendix A of 10CFR50, as described in Section 3.1 and in Subsection 7.3.2b.

(4) Extent of compliance to specific IEEE standards is discussed in Subsection 7.1.2.5.

(5) Extent of compliance to specific regulatory guides is discussed in Subsection 7.1.2.6.

7.1.2a.2 Mechanical Systems Separation Criteria

7.1.2a.2.1 General

(1) Separation of the affected mechanical systems and equipment is accomplished so that the substance and intent of the General Design Criteria of 10 CFR 50 Appendix A are fulfilled.

(2) Consideration is given to the redundant and diverse requirements of the affected systems.

(3) Consideration is given to the type, size, and orientation of possible breaks of the reactor coolant pressure boundary specified in Subsection 3.6.2.

(4) The protection afforded by the safety-related network satisfies the single active component failure criterion. A single active component failure is an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be part of the single failure. Fluid systems are considered to be designed against an assumed single failure if a single failure of any active component (assuming passive components function properly) does not result in a loss of capability of the system to perform its safety function.

(5) Redundant systems are separated from each other so that single failure of a component or channel will not interfere with the proper operation of its redundant/diverse counterpart.
The affected mechanical systems and equipment are separated so that systems important to safety are protected from the following hazards:

a) The pipe break dynamic effects outlined in Section 3.6

b) Environmental effects as a result of pipe breaks and as outlined in Section 3.11

c) Flooding effects as a result of pipe breaks and as outlined in Section 3.11

d) Missiles as defined in Section 3.5

e) Fires capable of damaging redundant mechanical safety equipment.

The need for and adequacy of separation to protect the safety equipment from the hazards are determined in conjunction with the criteria specified in Sections 3.4, 3.5, 3.6, 3.11, and 9.5.

7.1.2a.2.1.1 Separation Techniques

The methods used to protect redundant safety systems from the design basis hazards fall into four categories of separation techniques: plant arrangement, barriers, spatial separation, and alternatives.

a) Plant Arrangement

A basic design consideration of plant layout is that redundant divisions of a safety system should not share common equipment areas. However, equipment common to a particular safety system division can share a common area if that equipment does not constitute a hazard within itself to another safety system of the same division.

As an example, failure of a safety related pipe in Division I should not result in a failure of a pipe in Division II and vice versa.

Failure of any non-safety-related structure system or component shall not result in failure of any safety-related structures, system, or component.

To accomplish separations through plant arrangement, redundant divisions of a safety system may be placed in different compartments or even on different elevations. Non-safety equipment, components, or piping should not be run above safety equipment unless they are adequately restrained or it can be demonstrated that failure will not impair function of the safety equipment.

b) Barriers

Barriers are most often used in restricted areas where a particular hazard (e.g., small turbine missiles) is more easily identified or where other techniques are inappropriate (e.g., separation between control boards). Separation by barriers is an extension of separation by the use of compartments in plant arrangement.

Separation was also accomplished through the use of suitably designed equipment that in itself acts as a barrier. Examples would be heavily constructed control boards or heavy wall conduits and enclosed cable trays. In many cases, the barrier may enclose the hazard (e.g., a compartment around a high-speed turbine driven pump) in lieu of effecting a direct separation between redundant systems.

c) Spatial Separation

Spatial separation is another method of separating redundant safety systems and protecting them from the hazards described in Subsection 3.12.2.1.1.

For example, in areas where a barrier would be impractical, piping has been rerouted so that energy from a jet resulting from a break would be dissipated by the distance traveled. In this example, partial barriers or restraints could also be used, as well as by hardening design (e.g., heavier housing construction) of system components within the hazard area. When it can be shown that a hazard would have only a certain sphere of effectiveness (e.g., for pipe whip, a rotation about a plastic hinge at the next restraint), spatial separation is considered adequate.

d) Alternatives

When one of the above techniques is impractical, a suitable alternative is used, some of which are additional restraints, hardening design, or temporary system isolation under accident conditions. When the redundant safety component cannot be held safe from common hazards by the alternatives outlined above, more resistant components are selected. An example would be the use of high pressure piping in a low pressure safety system to ensure its ability to withstand the effect of a break in adjacent high pressure lines.

7.1.2a.2.2 System Separation Criteria

Piping for a redundant safety system is run independently of its counterparts, unless it can be shown that no single credible event, e.g., LOCA is capable of causing piping failure that could prevent reactor shutdown. Supports and restraints of redundant mechanical components and piping are not shared, unless such sharing does not significantly impair their ability to perform their safety function.

Penetrations to the primary containment are separated or other adequate provisions are made so that the initial break of one piping branch of a system does not render its redundant counterpart(s) inoperable.

7.1.2a.2.3 Physical Separation

(1) Mechanical equipment and piping are separated from each other so that single failure of a device or component will not interfere with the proper operation of its redundant/diverse counterpart.

(2) The ADS system is separated from the HPCI system such that no break location within the normally pressurized portion of the HPCI steam line can damage any component considered essential to the operation of either redundant division of ADS.

(3) The coolant injection portions of the ECCS are separated into the following functional groups:

a. HPCIS + 1 CSS + 1 LPCIS + with one RHR heat exchanger and 100% service water.

b. 1 CSS + 1 LPCIS + with one RHR heat exchanger and 100% service water.

(4) The equipment in each group is separated from that in the other group by the required practical distance. In addition, the HPCI and the RCIC systems are adequately separated.

(5) Separation barriers are constructed between the functional groups as required to assure that environmental disturbances (such as fire, flood, pipe rupture phenomena, falling objects, etc.) affecting one functional group will not affect the remaining groups. In addition, separation barriers are provided as required to assure that such disturbances do not affect both the RCIC and HPCI systems.

7.1.2a.3 Electrical Systems Separation Criteria

7.1.2a.3.1 General

(1) Separation of the affected electrical systems and equipment is accomplished so that the substance and intent of IEEE 279-1971, 10CRF50 Appendix A, General Design Criteria 3, 17 and 21 are fulfilled as further clarified and limited below.

(2) Consideration is given to the redundant and diverse requirements of the affected systems.

(3) The protection afforded by the safety-related network satisfies the single active component failure criterion. A single active component failure is an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be part of the single failure.

(4) Redundant systems are separated from each other so that single failure of a component or channel will not interfere with the proper operation of its redundant/diverse counterpart.

(5) The need for and adequacy of separation to protect the safety equipment from the above hazards are determined in conjunction with the criteria specified in Section 3.12.

(6) The affected electrical systems and equipment are separated so that systems important to safety are protected from the following hazards:

a) Fires in cable raceways due to an electrical fault that could cause failure of insulation of other cables.

b) Gross failure of electrical equipment in a single compartment of a control panel

c) Mechanical damage of electrical equipment in a single location.

d) Single Design Basis event, such as earthquake.

7.1.2a.3.1.1 Separation Techniques

The methods used to protect redundant safety systems from the design basis hazards fall into two categories of separation techniques: safety class structures, spatial separation.

a) Safety Class Structure

Basic design of plant layout is performed such that redundant circuits and equipment are located in separate areas or rooms insofar as practical.

The separation of Class 1E circuits and equipment are such that the required independence will not be compromised by the failure of mechanical systems served by the Class 1E electrical system. For example, Class 1E circuits are routed or protected such that failure of related mechanical equipment of one redundant system can not disable Class 1E circuits or equipment essential to the operation of the other redundant systems. The separation of Class 1E circuits and equipment make effective use of features inherent in the plant design such as using different rooms or opposite side of rooms or areas.

b) Spatial Separation

Spatial separation and/or protective barriers are such that no locally generated force or missile can destroy redundant RPS or ESF functions. In the absence of confirming analysis to support less stringent requirements, the following rules apply:

1. In rooms or compartments having heavy rotating machinery, such as the turbine-generator, or the reactor feedwater system pumps, or in rooms containing high-pressure feedwater piping, or high-pressure steam lines such as those between the reactor and the turbine, a minimum separation of 20 feet or a 6 inch thick reinforced concrete wall is required between trays containing cables of different divisions.

2. Any redundant switchgear, associated with two redundant RPS or ESF's and located in a potential mechanical damage zone such as discussed above, must have a minimum horizontal separation of 20 feet or must be separated by a protective wall equivalent to a 6 inch thick reinforced concrete wall.

3. In any compartment containing an operating crane, such as the turbine building main floor and the region above the reactor pressure vessel, there must be a minimum horizontal separation of 20 feet or a 6 inch thick reinforced concrete wall between trays containing cables from different divisions.

4. Plant area for spacial separation is discussed in Section 3.12.

5. Cable spreading room area for spacial separation is discussed in Section 3.12.

6. NSSS Main Control Room and Relay Room Panels

   a) The protection system and ESF control, relay and instrument panels/racks are located in safety class structures. Redundant protective systems and ESF panels and racks are located in different areas thus avoiding a potential source of missiles or pipe break energy release that could destroy redundant safety functions. Where this is not possible suitable barriers are provided between the panels/racks and potential missile and the effects of a pipe break.

   b) Control, relay and instrument panels/racks are designed in accordance with the following general criteria to preclude the possibility of fire propagating between redundant circuits whose loss would prevent safe shutdown of the plant:

      i. Single panels or instrument racks do not contain circuits or devices of the redundant protection system of ESF systems except:

         Certain operator interface control panels have operational considerations which dictate that redundant protection system or ESF system circuits or devices be located on a single panel. These circuits and devices are separated horizontally and vertically by a minimum distance of 6 inches or by steel barriers, or enclosures.

      ii. Certain panels contain redundant RPS or ESF circuits due to system requirements. In these panels the 6 inch separation required in panel wiring is implemented wherever possible. Where an exception to 6 inch separation exists, for example at logic relays accepting signals from more than one division, one of the two division's wiring is run in conduit. Another exception is the Neutron Monitoring system.

         These exceptions are covered by General Electric Licensing Topical Report NEDO-10139, "Compliance of Protection Systems to Industry Criteria: General Electric BWR Nuclear

Steam Supply System," dated June, 1970 which presents an analysis for each safety system relay panel bay containing wiring from redundant divisions used in developing the combinational logic for the design. This report was specifically prepared to describe panels for BWR/4 type plants. The analysis is performed for panels without barriers and without regard to spatial separation and evaluates the loss of an entire safety system relay panel bay considering all combinations of open circuit and short circuit to either ground or power circuits. The report concludes that no safety functions are prevented following complete loss of such a bay.

iii. Where the above separation methods (i) or (ii) are not feasible, one of the separation group circuits are to be covered with a qualified non-metallic barrier material. A description of the materials and analysis to regulatory requirements is provided in Subsection 3.13 (conformance to Regulatory Guide 1.75).

iv. If two panels containing circuits of different separation divisions are less than 3 feet apart, there is a steel barrier between the two panels. Panel ends closed by steel end plates are considered to be acceptable barriers provided that terminal boards and wireways are spaced a minimum of one inch from the end plate.

v. Panel-to-floor fireproof barriers are provided between adjacent panels of different divisions, and divisional equipment on the same panel.

vi. Penetration of separation barriers within a subdivided panel is permitted, provided that such penetrations are sealed or otherwise treated so that fire generated by an electrical fault could not reasonably propagate from one section to the other and disable a protective function.

7.1.2a.3.2 Identification

Major electrical equipment of safety-related systems shall be identified so that two facts are physically apparent to operating and maintenance personnel: first, the equipment is part of the RPS or ESF equipment; and second, the grouping (or division) of enforced segregation with which the equipment is associated. Identification and division assignment conform to the following:

(1) Panels and Racks

Panels and racks associated with the RPS or ESF shall be labeled with marker plates which are conspicuously different from those for other similar panels; the difference may be in color, shape, or color of engraving-fill. The marker plates include identification of the proper division of the equipment as listed in Table 3.12-1.

(2) Junction or Pull Boxes

Junction and/or pull boxes enclosing wiring for the RPS or ESF have identification similar to and compatible with the panels and racks considered above.

(3) Cables

Cables external to cabinets and/or panels for the RPS or ESF are marked to distinguish them from other cables and identify their separation division as applicable. This identification requirement does not apply to individual conductors.

(4) Raceways

Those trays or conduits which carry RPS or ESF wiring are identified at entrance points of each room through which they pass (and exit points unless the room is small enough to facilitate convenient following of cable) with a permanent marker identifying their assigned division.

(5) Sensory Equipment Grouping and Designation Letters

Redundant sensory equipment for RPS or ESF are identified by suffix letters in accordance with Table 7.1-10 for RPS, and other deenergize to operate systems, Table 7.1-12 for ECCS, RCIC and other energize to operate systems and Table 7.1-11 for the Neutron Monitoring System. These tables also show the allocation of sensors to their separated divisions.

(6) PGCC Cables and Raceways

Cables and raceways are marked at the entrance to and exit from the PGCC floor sections and at 10 foot intervals within the floor sections. Class 1E raceways within the PGCC floor sections are identified prior to the installation of their cables.

The marking device is a permanent color coded band to distinguish between redundant Class 1E cables and non-Class 1E cables.

7.1.2a.3.3 System Separation Requirements

7.1.2a.3.3.1 Reactor Protection System (RPS)

(1) RPS cables are run in rigid or flexible metal conduits (if the conduits are qualified as a short circuit protection barrier able to carry ground current of 30 amps for 30 seconds) except 3 out of 4 channels of the RPS sensing circuits, located in the turbine building, are routed through a short length (40'-100') PVC embedded conduits. A ground fault return conductor capable of carrying 30 amps for 30 seconds is routed with each embedded conduits. Conduits or raceways containing neutron monitoring sensor cables (SRM, IRM, and LPRM) must be run in their own covered raceways which contain no other RPS cables. Cable installation beneath the reactor vessel is described in Section 8.1.6.1 (Regulatory Guide 1.75 (1/75), Part 15).

(2) Wiring to duplicate sensors on a common process tap is run in separate raceways to their separate destinations even though there is a functionally redundant set of sensors on a redundant process tap.

(3) Wiring for sensors of more than one variable in the same trip channel may be run in the same divisional raceway.

(4) Wires from the RPS trip system to a single group of scram solenoids may be run in a single raceway. A single raceway does not contain wires to more than one group of scram solenoids. RPS raceways contain only RPS wires.

(5) Cables through the primary containment penetrations are so grouped that failure of all cabling in a single penetration cannot prevent a scram. (This applies also to the neutron monitoring sensor cables and the main steam isolation valves position switch cables.)

(6) Power supplies to systems which de-energize to operate (so called "fail-safe" power supplies) require only that separation which is deemed prudent to assure availability. Therefore, the protection system fly-wheel motor generator (MG) sets, load circuit breakers and power wiring are not required to comply with these separation requirements event though the power is wired to separated panels.

(7) The RPS has a minimum of four independent input instrument channels for each measured variable.

(8) The RPS wiring is run and/or protected such that no common source of potentially damaging energy (e.g., electrical fire in non-RPS wireways, malfunction of plant equipment, pipe rupture, etc.) could reasonably result in loss of ability to scram when required.

7.1.2a.3.3.2 Emergency Core Cooling System (ECCS) and Nuclear Steam Supply Shutoff System (NSSSS)

(1) Separation is such that no single failure can prevent operation of an engineered safeguard function. Redundant (even dissimilar) systems may be required to perform the required function to satisfy the single failure criterion. Nuclear Steam

Supply Shutoff System fail-safe circuits which de-energize to operate follow the cable separation requirements described in Subsection 7.1.2a.3.3.1.

(2) The inboard and outboard NSSSS isolation valves are backups for each other so they must be independent of and protected from each other to the extent that no single failure can prevent the operation of at least one of an inboard/outboard pair.

(3) Isolation valve circuits require special attention because of their function in limiting the consequences of a pipe break outside the primary containment. Isolation valve control and power circuits shall be protected from the pipe lines that they are responsible for isolating as follows:

a. Essential isolation valve wiring in the vicinity of the outboard valve (or downstream of the valve) is installed in conduit and routed such as to take advantage of the mechanical protection afforded by the valve operator or other available structural barriers not susceptible to disabling damage from the pipe line break. Additional mechanical protection (barriers) are interposed as necessary.

b. Divisional Assignment

MOVs which have a mechanical check valve backup for their isolation function are included in the division which embraces the system in which the valves are located rather than adhering strictly to the inboard/outboard divisional classification.

7.1.2a.3.3.3 Steam Leakage Zone

Electrical equipment and raceways for systems listed in Subsection 7.1.2a.2.3 avoid location in a steam leakage zone insofar as practical, or are designed for short-term exposure to the high temperature and humidity associated with a steam leak.

7.1.2a.3.3.4 Suppression Pool Level Swell Zone

Any electrical equipment and/or raceways for RPS or ESF located in this zone are designed to satisfactorily complete their function before being rendered inoperable due to exposure to the environment created by the level swell phenomena.

7.1.2a.3.3.5 Penetrations

Penetrations are so arranged that no design basis-event can disable cabling in more than one penetration assembly. Penetrations contain cables of one divisional assignment only.

7.1.2a.3.3.6 Power Generation Control Complex - (PGCC)

Detailed description and safety evaluation aspects for a typical PGCC System are presented in GE-Topical Report: "Power Generation Control Complex; NEDO-10466A" and its amendments.

The PGCC rooms include control panels, connectors, floor sections, and termination cabinets. The floor sections are divided into ducts and the termination cabinets have metallic barriers to separate redundant Class 1E wiring.

The floor section ducts are designed so that each duct acts as a raceway and has adequate fire barriers and will contain wiring of only one redundant circuit. The ducts have solid metal walls and floor and a removable solid metal cover.

7.1.2a.3.3.7 Annunciator and Computer

All annunciator and computer input circuits are classified as non-Class 1E circuits. The cable runs of these circuits are separated from Class 1E circuits by the minimum separation requirements specified in Section 3.12. However, these non-Class 1E circuits are not separated from Class 1E control circuits within 1E panels in which the non-Class 1E circuit derives its input, (e.g., circuit breaker auxiliary contact used for computer input) or within the PGCC assembly. These non-Class 1E instrument circuits are considered to be low energy and the probability of these non-Class 1E circuits providing a mechanism of failure to the Class 1E circuits is extremely low.

7.1.2a.3.3.8 Conformance to IEEE 384-1974, 1981

The safety-related systems described in Sections 7.2, 7.3, 7.4, and 7.6 meet the independence and separation criteria for redundant systems in accordance with IEEE 279, paragraph 4.6. IEEE 384 is not applicable to the original plant, however, the following features are provided:

The electrical power supply, instrumentation, and control wiring for redundant portions of safety related systems have physical separation to preserve redundance and ensure that no single credible event will prevent operation of the associated function. Credible events include, but are not limited to, the effects of short circuits, pipe rupture, pipe whip, high pressure jets, missiles, fire, earthquake, and falling objects, and are considered in the basic plant design.

The independence of tubing, piping, and control devices for safety-related controls and instrumentation is achieved by physical space or barriers between separation groups of the same protective function. In locations where a specific hazard exists (missile, jet, etc.) which could produce damage to safety-related controls and instrumentation, the physical separation or structural protection provided will be adequate to ensure that no multiple failures can result from a single common event.

IEEE 384-1981 is applicable to the safety-related systems in the Diesel Generator 'E' Facility.

The criteria and bases for the independence of electrical cable, including routing, marking and cable derating, are covered in Section 8.3, 7.1.2a.3.2(6) and 7.1.2a.3.3.6. Fire detection and protection in the areas where wiring is installed is covered in Subsections 9.5.1 and 7.1.2a.3.3.6.

7.1.2a.4 NSSS Instrument Errors

The design considers instrument drift, setability and repeatability in the selection of instrumentation and controls and in the determination of setpoints is provided to allow for instrument error. The safety limits and allowable values are listed in the plant Technical Specifications. The trip setpoints are listed in the Technical Requirements Manual. The amount of instrument error is determined by test and experience. The setpoint is selected based on the known error. The test frequency is greater on instrumentation that demonstrates a tendency to err.

7.1.2b.4 Non-NSSS Instrument Errors

Consideration of NSSS instrument errors applies to non-NSSS LOCA signal and diesel start. See Section 7.6 for discussion and references to NSSS sections which discuss initiation in core spray and RHR nuclear steam supply shutoff systems.

7.1.2.5 Conformance to Industry Standards

This section covers both NSSS and non-NSSS application of standards. Statements of conformance are indicated as to NSSS and/or non-NSSS applicability.

On June 6, 1987, a fifth diesel generator designated Diesel Generator 'E' was added to the standby power system as part of the onsite power system. The modification that added Diesel Generator 'E' was based on applicable codes and standards in effect on September 22, 1983. These later codes and standards are only applicable to the Diesel Generator E building and the modifications in the Diesel Generator A, B, C and D rooms to add the transfer points and interconnections.

7.1.2.5.1 Conformance to IEEE 279-1971

This discussion is presented on a system-by-system basis in the analysis portions of Sections 7.2, 7.3, 7.4, and 7.6.

7.1.2.5.2 Conformance to IEEE 308-1974

Conformance to IEEE 308-1974 is described in Section 8.1.6.1 (Regulatory Guide 1.9 and 1.32).

7.1.2.5.2.1 Conformance to IEEE 308-1980

The Diesel Generator 'E' Facility is designed in accordance with IEEE 308-1980.

7.1.2.5.3 Conformance to IEEE 323-1971

a) NSSS - Compliance with IEEE 323-1971: written procedures and responsibilities are developed for the design and qualification of all Class I electric equipment. This includes preparation of specifications, qualification procedures, and documentation for both NSSS supplies manufactured and NSSS supplies purchased Class I equipment. Qualification testing or analysis is accomplished prior to release of the engineering design for production. Standards manuals are maintained containing specifications, practices, and procedures for implementing qualification requirements, and an auditable file of qualification documents is available for review. See the Susquehanna SES Environmental Equipment Qualification Program.

b) Non-NSSS - See the Susquehanna SES Environmental Equipment Qualification Program.

7.1.2.5.3.1 Conformance to IEEE 323-1974

The Diesel Generator 'E' Facility equipment meets the requirements of IEEE 323-1974 as it applies to the mild environment in the diesel generator buildings.

7.1.2.5.4 Conformance to IEEE 336-1971

a) NSSS – Specifications include requirements for conformance to IEEE 336.

b) Non-NSSS refer to Section 3.13 for Regulatory Guide 1.30.

7.1.2.5.4.1 Conformance to IEEE 336-1980

The Diesel Generator 'E' Facility equipment meets the requirements of IEEE 336-1980. The Diesel Generator 'E' equipment is installed, inspected, and tested under the requirements of the Operational Quality Assurance Program as described in Section 17.2.

7.1.2.5.5 Conformance to IEEE 338-1971

This discussion is presented on a system by system basis in the analysis portion of Sections 7.2, 7.3.2a, 7.3.2b, 7.4, and 7.6.

7.1.2.5.5.1 Conformance to IEEE 338-1977

The Diesel Generator 'E' system and equipment meet the requirements of IEEE 338-1977.

7.1.2.5.6 Conformance to IEEE 344-1971

a) NSSS - All safety-related instrumentation and control equipment is classified as Seismic Category I, designed to withstand the effects of the safe shutdown earthquake (SSE) and remain functional during normal and accident conditions. Qualification and documentation procedures used for Seismic Category I equipment and systems meet the provisions of IEEE 344 as identified in Section 3.10a. See the Susquehanna SES Environmental Equipment Qualification Program.

b) Non-NSSS - See Subsection 8.1.6.2, Sections 3.10b and 3.10c. See the Susquehanna SES Environmental Equipment Qualification Program.

c) For those systems identified in Table 7.1-3, the supplemental requirements of Branch Technical Position EICSB 10, Electrical and Mechanical Equipment Seismic Qualification Program are applicable.

7.1.2.5.6.1 Conformance to IEEE 344-1975

The Diesel Generator 'E' equipment meets the requirements of IEEE 344-1975 as it applies to the diesel generator buildings. See Subsection 3.10b and 3.10c.

7.1.2.5.7 Conformance to IEEE 379-1972

a) NSSS - The extent to which the single failure criteria of IEEE 379 is satisfied is specifically covered in the analysis of each system to the requirements of IEEE 279, paragraph 4.2 - see Subsection 7.1.2.5.1.

b) Non-NSSS - See the analysis of systems to meet requirements of IEEE 279, paragraph 4.2 in Subsection 7.3.2b.2.

7.1.2.5.7.1 Conformance to IEEE 379-1977

The Diesel Generator 'E' system and equipment are designed so that whenever the Diesel Generator 'E' is aligned to the Class 1E electrical system, the Class 1E electrical system still meets the single failure criteria as discussed in paragraph 4.2 in Subsection 7.3.2b.2.

7.1.2.5.8 Conformance to IEEE 384-1974

a) NSSS - Refer to Subsection 7.1.2a.3.3.8.

b) Non-NSSS - Refer to Section 3.12.

7.1.2.5.8.1 Conformance to IEEE 384-1981

The Diesel Generator 'E' equipment meets the requirements of IEEE 384-1981.

7.1.2.6 Conformance to Regulatory Guides

This section covers both NSSS and non-NSSS application of regulatory guides. Statements of conformance are indicated as to NSSS and/or non-NSSS applicability.

7.1.2.6.1 Conformance to Regulatory Guide 1.6 (3/10/71)

Refer to Subsection 8.1.6.1, paragraph a and Section 3.13.

7.1.2.6.2 Conformance to Regulatory Guide 1.9 (5/10/71)

Refer to Section 3.13 and Subsection 8.1.6.1, paragraph b.

7.1.2.6.2.1 Conformance to Regulatory Guide 1.9 (December 1, 1979)

For conformance of the Diesel Generator 'E' equipment to Regulatory Guide 1.9 refer to Section 3.13.

7.1.2.6.3 Conformance to Regulatory Guide 1.11 (3/10/71)

Refer to Section 3.13.

7.1.2.6.4 Conformance to Regulatory Guide 1.22 (2/17/72)

This discussion is presented for systems in the analysis portion of Sections 7.2, 7.3, 7.4, and 7.6.

7.1.2.6.5 Conformance to Regulatory Guide 1.29 (6/7/72)

a) NSSS - The instrumentation and control equipment required to meet Seismic Class I by Regulatory Guide 1.29 is identified in Table 3.2-1.

b) Non-NSSS - Refer to Section 3.13.

7.1.2.6.5.1 Conformance to Regulatory Guide 1.29 (9/78)

The conformance of the Diesel Generator 'E' equipment with Regulatory Guide 1.29 is discussed in Section 3.13.

7.1.2.6.6 Conformance to Regulatory Guide 1.30 (8/11/72)

a) NSSS - The quality assurance requirements of IEEE 336-1971 are applicable during the plant design and construction phases (see Subsection 7.1.2.5.5) and will also be implemented as an operational QA program during plant operation in response to Regulatory Guide 1.30.

b) Non-NSSS - Refer to Section 3.13.

7.1.2.6.7 Conformance to Regulatory Guide 1.32 (8/72)

a) NSSS - The ECCS is designed to the requirements of Regulatory Guide 1.32 and IEEE Standard 308-1971. Subsection 7.3.2 provides discussion of compliance.
b) Non-NSSS – Refer to Sections 3.13 and Section 8.1.6.1.

7.1.2.6.7.1 Conformance to Regulatory Guide 1.32 (2/77)

The conformance of the Diesel Generator 'E' equipment with Regulatory Guide 1.32 is discussed in Section 3.13.

7.1.2.6.8 Conformance to Regulatory Guide 1.40 (3/16/73)

a) NSSS - There are no continuous duty motors installed inside the containment that are part of the instrumentation and control systems and no discussion is provided.

b) Non-NSSS – Refer to Subsection 3.11.2 and Section 3.13.

7.1.2.6.9 Conformance to Regulatory Guide 1.45 (5/73)

Refer to Subsections 3.13 and 5.2.5.1 for detailed description of the Susquehanna SES design conformance to this guide.

7.1.2.6.10 Conformance to Regulatory Guide 1.47 (5/73)

a) NSSS - The system of bypass indication is designed to satisfy the requirement of IEEE 279-1971 paragraph 4.13 and Regulatory Guide 1.47 and is discussed for each safety-related system under Sections 7.2, 7.3, 7.4, and 7.6. The design of the bypass indication system allows testing during normal operation and is used to supplement administrative procedures by providing indications of safety systems status.

The bypass indication system is designed and installed in a manner which precludes the possibility of adverse affects on the plant safety system. The bypass indication system is electrically isolated from the protection circuits such that the failure or bypass of a protective function is not a credible consequence of failures in the bypass indication system and the bypass indication system cannot reduce the independence between redundant safety systems.

b) Non-NSSS - Refer to individual systems in Section 7.3 and discussion in Section 7.5.

7.1.2.6.11 Conformance to Regulatory Guide 1.53 (6/73)

a) NSSS - The safety-related system designs conform to the single failure criterion. The analysis portions of Sections 7.2, 7.3, 7.4 and 7.6 provide further discussion.

b) Non-NSSS Refer to Section 3.13

7.1.2.6.12 Conformance to Regulatory Guide 1.62 (10/73)

a) NSSS - Manual initiation of the protective action is provided at the system level in the Reactor Protection System, (primary) Containment and Reactor Vessel Isolation Control System and Emergency Core Cooling Systems. The analysis portions of Sections 7.2 and 7.3 provide further discussion.

b) Non-NSSS - Refer to Section 3.13.

7.1.2.6.13 Conformance to Regulatory Guide 1.63 (10/73)

a) NSSS - Regulatory Guide 1.63 applies to electrical penetration assemblies which are not part of NSSS scope.

b) Non-NSSS - Refer to Section 3.13.

7.1.2.6.14 Conformance to Regulatory Guide 1.68 (11/73)

Refer to Section 3.13.

7.1.2.6.15 Conformance to Regulatory Guide 1.70 (Rev. 2)

The format and content of Chapter 7 conform to the requirements of Regulatory Guide 1.70.

7.1.2.6.16 Conformance to Regulatory Guide 1.73 (1/74)

Refer to Section 3.13.

7.1.2.6.17 Conformance to Regulatory Guide 1.75 (1/75)

a) NSSS Regulatory Guide 1.75 is not applicable to Susquehanna SES; however, degree of compliance to separation criteria of IEEE 384 is discussed in Subsection 7.1.2.5.8.

b) Non-NSSS - Refer to Section 3.13 and Subsection 8.1.6.1.

7.1.2.6.17.1 Conformance to Regulatory Guide 1.75 (9/78)

The conformance of the Diesel Generator 'E' equipment with Regulatory Guide 1.75 is discussed in Section 8.1.6.1.

7.1.2.6.18 Conformance to Regulatory Guide 1.80 (6/74)

a) NSSS - Regulatory Guide 1.80 applies to the testing of instrument air systems which are not part of the NSSS scope.

b) Non-NSSS - Refer to Section 3.13.

7.1.2.6.19 Conformance to Regulatory Guide 1.89 (11/74)

a) NSSS - See the Susquehanna SES Environmental Equipment Qualification Program.

b) Non-NSSS - Refer to Section 3.13.

7.1.2.6.20 Conformance to Regulatory Guide 1.96 (6/76)-Rev. 1

The Main Steam Isolation Valve Leakage Control System has been deleted. The function is now performed by the Isolated Condenser Treatment Method (Section 6.7), approved by the NRC as an alternative to Regulatory Guide 1.96.

7.1.2.6.21 Conformance to Regulatory Guide 1.97

Post-accident instrumentation is in conformance with Regulatory Guide 1.97, Revision 2, with clarifications as described in PLA-965 and PLA-2222. Equipment and components used for post-accident monitoring are described in the applicable FSAR sections.

The redundant valve position indication in the control room as discussed in PLA-2222, Conformance to Regulatory Guide 1.97 Revision 2, is not applicable to the following solenoid operated Primary Containment Isolation Valves (PCIV) belonging to the Post-accident H_20_2 Analyzers subsystem:

Unit-1 PCIVs: SV15742A(B), SV15774A(B), SV15752A(B), SV15782A(B), SV15734A(B), SV15740A(B), SV15776A(B), SV15750A(B), SV15780A(B), SV15736A(B),
Unit-2 PCIVs: SV25742A(B), SV25774A(B), SV25752A(B), SV25782A(B), SV25734A(B), SV25740A(B), SV25776A(B), SV25750A(B), SV25780A(B), SV25736A(B).

The inboard and outboard PCIVs and their associated position indication circuitry are powered from the same division of class-1E power as their post-accident H_20_2 analyzers. The basis for this design is to prevent a single failure of one power source from eliminating all post accident monitoring capability. The post-accident H_20_2 analyzer piping is a closed system. The closed system provides a redundant isolation barrier for the primary containment. It is a passive isolation barrier that does not require any position indication. The design was determined acceptable in the NRC Safety Evaluation related to the Technical Specification amendment 170 and 195. (PPL document reference NRC 2001-0113).

All five valves of a group are manually operated with one control switch located in the control room. They are designed to open or close in a group and can not be opened without their power sources. Each group is provided with one set of valve position indicating lights in the control room. An amber light in the control room indicates that all valves in a given group are closed. A red light in the control room indicates that all valves in that group are open. Dual indication (red and amber both illuminated) or lack of indication (neither light illuminated) indicates a problem. Individual valve position indicating lights are provided on local control panels. They provide additional means to determine the positions of these PCIVs.

7.1.2.7 Technical Design Bases

The technical design bases for RPS are in Subsection 7.2.1, for engineered safety features in Subsection 7.3.1, for systems required for safe shutdown in Subsection 7.4.1, and for other systems required for safety in Subsection 7.6.1a.

7.1.2.8 Safety System Settings

The safety system setpoints are listed in the Technical Specifications. The settings are determined based on operating experience and conservative analyses. The settings are high enough to preclude inadvertent initiation of the safety action, but low enough to assure that significant margin is maintained between the actual setting and the limiting safety system settings. Instrument drift, setability and repeatability are considered in the setpoint determination (see Subsections 7.1.2a.4 and 7.1.2b.4). The margin between the limiting safety system settings and the actual safety limits include consideration of the maximum credible transient in the process being measured.

The periodic test frequency for each variable is determined from experimental data on setpoint drift and from quantitative reliability requirements for each system and its components.

TABLE 7.1-1

RESPONSIBILITY

	NSSS	Non-NSSS
Reactor Trip System		
Reactor Protection System (RPS)	X	
Alternate Rod Injection System (ARI)		X
Engineered Safety Feature Systems		
Emergency Core Cooling Systems (ECCS)	X	
High Pressure Coolant Injection (HPCI)		
Automatic Depressurization System (ADS)		
Core Spray System (CS)		
Low Pressure Coolant Injection (LPCI) Mode of RHR		
Primary Containment and Reactor Vessel		
Isolation Control Systems (PCBVICS)	X	X
RHR, Containment Spray Cooling Mode	X	
RHR, Suppression Pool Cooling Mode	X	
PCRVICS Leak Detection System	X	
PCRVICS Radiation Monitoring Systems	X	
Primary Containment Isolation Controls		X
Combustible Gas Control System		X
Primary Containment Vacuum Relief		X
Standby Gas Treatment System (SGTS)		X
Reactor Building Recirculation System		X
Reactor Building Isolation and HVAC Support		X
Habitability Systems, Control Room Isolation and Supporting HVAC Systems		
Control Room HVAC		X
Control Structure HVAC		X
Computer Room Cooling System		X
Emergency Outside Air Supply		X
Battery Room Exhaust System		X
Auxiliary Support Systems		
Emergency Service Water (ESW)		X
RHR Service Water (RHRSW)		X
Containment Instrument Gas		X
Standby Power Systems		X
Heating, Ventilating, and Air Conditioning for ESF Areas		X

TABLE 7.1-1

RESPONSIBILITY

	NSSS	Non-NSSS
Instrumentation and Controls for Systems Required for Safe Shutdown		
Reactor Core Isolation Cooling System (RCIC)	X	
Standby Liquid Control System (SLC)	X	
RHR, Reactor Shutdown Cooling Mode	X	
Reactor Shutdown Outside the Control Room		X
Safety Related Display Instrumentation	X	X
Other Systems Required for Safety		
High Pressure/Low Pressure Interlocks	X	
NSSS Leak Detection System	X	
Neutron Monitoring System		
Intermediate Range Monitor (IRM)	X	
Average Power Range Monitor (APRM)	X	
Local Power Range Monitor (LPRM)	X	
Oscillating Power Range Monitor (OPRM)	X	
Recirculation Pump Trip System	X	
Drywell Entry Purge (Air Purge)		X
Containment Atmosphere Monitor		X
NSSS to non-NSSS (Standby Power Start)	X	X
Process Effluent Radiological Monitoring	X	X
Control Systems Not Required for Safety		
Refueling Interlocks	X	
Reactor Vessel Instrumentation	X	
Reactor Manual Control System	X	
Rod Block Trip Subsystem	X	
Rod Worth Minimizer Subsystem	X	
Recirculation Flow Control System	X	
Feedwater Control System	X	X
Pressure Regulator & Turbine-Generator System		X
Neutron Monitoring System	X	
Traversing Incore Probe (TIP)		
Rod Block Monitor (RBM)	X	
Source Range Monitor (SRM)		
Reactor Water Cleanup System (RWCU)	X	X
Radwaste System		X
Gaseous Radwaste System		
Liquid Radwaste System		
Solid Radwaste System		
Area Radiation Monitoring System	X	X
Process Computer	X	

HISTORICAL INFORMATION

TABLE 7.1-2

SIMILARITY TO LICENSED REACTORS

	Instrumentation and Control (System)	Plants Applying for or Having Construction Permit or Operating License	Similarity of Design
(1)	Reactor Protection System	Shoreham	See Note 1
(2)	Primary Containment and Reactor Vessel Isolation Control System	Shoreham	Identical See Note 9
(3)	Emergency Core Cooling System	Peach Bottom 2 & 3	Similar
(4)	Neutron Monitoring System	Shoreham	Identical except more incore sensors
(5)	Refueling Interlocks	LaSalle	Identical
(6)	Reactor Manual Control System	Shoreham	Identical
(7)	Reactor Vessel-Instrumentation	Fermi 2	Similar See Note 3
(8)	Recirculation Flow Control System	Shoreham	Identical
(9)	Feedwater Control System	Peach Bottom 2 & 3	See Note 6
(10)	Process Radiation Monitoring Equipment	Hatch 1	See Note 2
(11)	Area Radiation Monitoring Equipment	Hatch 1	Identical
(12)	Process Computer	None	New
(13)	Reactor Core Isolation Cooling System	Shoreham	See Note 8
(14)	Standby Liquid Control System	Shoreham	Identical
(15)	Reactor Water Cleanup System	Shoreham	See Note 4
(16)	Leak Detection Systems (NSSS)	Hatch 2	See Note 5
(17)	RHR, Reactor Shutdown Cooling Mode	Shoreham	Identical
(18)	Main Steamline Isolation Valve Leakage Control	Shoreham	Identical
(19)	NSSS Safety-Related Display	Brunswick	Identical
(20)	RHR, Containment Spray Cooling Mode	Hatch 2	Identical

HISTORICAL INFORMATION

TABLE 7.1-2

SIMILARITY TO LICENSED REACTORS

Instrumentation and Control (System)	Plants Applying for or Having Construction Permit or Operating License	Similarity of Design
(21) Recirculation Pump Trip (RPT) System	LaSalle	See Note 7
(22) RHR, Suppression Pool Cooling Mode	Hatch 2	Identical

Note 1:

This plant has more control rods and a larger CRD scram discharge volume than Shoreham. It has sufficient discharge volume capacity to contain the required number of scrams and sufficient scram initiation equipment (control rod solenoids, actuator logic and control cables) to effect a scram.

Note 2:

The main steamline radiation monitoring subsystem is identical to Hatch 1.

Note 3: Reactor Vessel Instrumentation

Susquehanna has an upset level water range. Fermi does not.

Note 4:

Identical except for additional leak detection measurement on the RWCU inlet line. The signal is used to close the RWCU isolation valves.

Note 5:

Some variables are recorded on Susquehanna whereas they are indicated on Hatch 2.

TABLE 7.1-2

SIMILARITY TO LICENSED REACTORS

Instrumentation and Control (System)	Plants Applying for or Having Construction Permit or Operating License	Similarity of Design

Note 6:

Susquehanna SES uses GE MAC 7000 instrument loop whereas Peach Bottom 2 and 3 utilize GE MAC 5000.

Note 7:

The Susquehanna design requires that circuit breakers be located in an area where they are not exposed to environmental conditions that would cause them to fail to meet the interrupting time requirement.

Note 8:

Identical, except on Susquehanna

a) the outboard MSIV is normally open and has no warmup valve,
b) the inboard MSIV warmup valve is normally closed,
c) these are two turbine exhaust vacuum breaker valves.

Note 9:

The term "identical" means the referenced systems' instruments, controls and logics are functionally the same, although the sizes, flows, and locations may be different.

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TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX (3)

	10 CFR 50.34	10 CFR 50.36	10 CFR 50.55a	10 CFR 50 App. A GDC 1	10 CFR 50 App. A GDC 2	10 CFR 50 App. A GDC 3	10 CFR 50 App. A GDC 4	10 CFR 50 App. A GDC 5	10 CFR 50 App. A GDC 10	10 CFR 50 App. A GDC 12	10 CFR 50 App. A GDC 13	10 CFR 50 App. A GDC 15
RPS	X	X	X	X	X	X	X	X	X	X	X	X
PCRVICS	X	X	X	X	X	X	X		X		X	
ECCS	X	X	X	X	X	X	X		X		X	
NMS	X	X	X	X	X	X	X		X	X	X	
Reactor Manual Control Svstem	X	X		X	X	X	X				X	
Reactor Vessel Instrumentation	X		X								X	
Recirculation Flow Control	X		X								X	X
Feedwater Control System	X		X								X	X

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TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX [3]

	10 CFR 50.62	10 CFR 50.34	10 CFR 50.36	10 CFR 50.55a	10 CFR 50 App. A GDC 1	10 CFR 50 App. A GDC 2	10 CFR 50 App. A GDC 3	10 CFR 50 App. A GDC 4	10 CFR 50 App. A GDC 5	10 CFR 50 App. A GDC 10	10 CFR 50 App. A GDC 12	10 CFR 50 App. A GDC 13	10 CFR 50 App. A GDC 15
Process Computer		X		X									
RCIC		X	X	X	X	X	X	X		X		X	
Standby Liquid Control System	X	X	X	X	X	X	X	X				X	
Reactor Water Cleanup System		X		X								X	X
NSSS Leak Detection Systems		X	X	X	X	X	X	X		X		X	
Reactor Shutdown Cooling Mode (RHR)		X	X	X	X	X	X	X		X		X	
Alternate Rod Injection System (ARI)	X	X			X	X	X	X				X	

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TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX (3)

	10 CFR 50.34	10 CFR 50.36	10 CFR 50.55a	10 CFR 50 App. A GDC 1	10 CFR 50 App. A GDC 2	10 CFR 50 App. A GDC 3	10 CFR 50 App. A GDC 4	10 CFR 50 App. A GDC 5	10 CFR 50 App. A GDC 10	10 CFR 50 App. A GDC 12	10 CFR 50 App. A GDC 13	10 CFR 50 App. A GDC 15
Safety Related Display Instrumentation	X	X	X	X	X	X	X			X	X	X
Drywell Vacuum Relief System	X		X									
RHRS Containment Spray Cooling Mode	X	X	X	X	X	X	X				X	
Recirculation Pump Trip	X	X	X	X	X	X	X				X	
RHRS Suppression Pool Cooling Mode	X	X	X	X	X	X	X				X	

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TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX (3)

	10 CFR 50 App. A GDC 19	10 CFR 50 App. A GDC 20.	10 CFR 50 App. A GDC 21	10 CFR 50 App. A GDC 22	10 CFR 50 App. A GDC 23	10 CFR 50 App. A GDC 24	10 CFR 50 App. A GDC 25	10 CFR 50 App. A GDC 26	10 CFR 50 App. A GDC 27	10 CFR 50 App. A GDC 28	10 CFR 50 App. A GDC 29	10 CFR 50 App. A GDC 33	10 CFR 50 App. A GDC 34
RPS	X	X	X	X	X	X	X				X		
PCRVICS	X	X	X	X	X	X					X		
ECCS	X	X	X	X	X	X					X		X
NMS	X	X	X	X	X	X					X		
Reactor Manual Control Systems	X	X				X		X	X	X	X		
Reactor Vessel Instrumentation	X					X					X		
Recirculation Flow Control	X					X		X			X		
Feedwater Control System	X					X					X		

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TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX [3]

	10 CFR 50 App. A GDC 19	10 CFR 50 App. A GDC 20.	10 CFR 50 App. A GDC 21	10 CFR 50 App. A GDC 22	10 CFR 50 App. A GDC 23	10 CFR 50 App. A GDC 24	10 CFR 50 App. A GDC 25	10 CFR 50 App. A GDC 26	10 CFR 50 App. A GDC 27	10 CFR 50 App. A GDC 28	10 CFR 50 App. A GDC 29	10 CFR 50 App. A GDC 30	10 CFR 50 App. A GDC 33	10 CFR 50 App. A GDC 34
Process Computer	X										X			
RCIC	X	X	X	X	X	X					X			X
Standby Liquid Control System	X										X			
Reactor Water Cleanup System	X													
NSSS Leak Detection Systems	X	X	X	X	X	X					X	X	X	X
Reactor Shutdown Cooling Mode (RHR)	X	X	X	X	X	X					X			X
Spent Fuel Pool Cooling and Cleanup System														
Alternate Rod Injection System (ARI)	X	X	X	X		X					X			

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CODES AND STANDARDS APPLICABILITY MATRIX (3)

	10 CFR 50 App. A GDC 19	10 CFR 50 App. A GDC 20	10 CFR 50 App. A GDC 21	10 CFR 50 App. A GDC 22	10 CFR 50 App. A GDC 23	10 CFR 50 App. A GDC 24	10 CFR 50 App. A GDC 25	10 CFR 50 App. A GDC 26	10 CFR 50 App. A GDC 27	10 CFR 50 App. A GDC 28	10 CFR 50 App. A GDC 29	10 CFR 50 App. A GDC 33	10 CFR 50 App. A GDC 34
Safety Related Display Instrumentation	X					X				X	X	X	X
RHRS Containment Spray Cooling Mode	X		X	X	X	X					X		X
Recirculation Pump Trip		X	X	X	X	X					X		
RHRS Suppression Trip Cooling System	X		X			X					X		

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TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX [3]

	10 CFR 50 App. A GDC 35	10 CFR 50 App. A GDC 37	10 CFR 50 App. A GDC 38	10 CFR 50 App. A GDC 40	10 CFR 50 App. A GDC 41	10 CFR 50 App. A GDC 43	10 CFR 50 App. A GDC 44	10 CFR 50 App. A GDC 46	10 CFR 50 App. A GDC 50	10 CFR 50 App. A GDC 54	10 CFR 50 App. A GDC 55	10 CFR 50 App. A GDC 56	10 CFR 50 App. A GDC 57
RPS													
PCRVICS	X									X	X	X	
ECCS	X	X											
NMS													
Reactor Manual Control Svstem													
Reactor Vessel Instrumentation													
Recirculation Flow Control													
Feedwater Control System													
Process Computer													
RCIC													
Standbv Liauid Control Svstem													

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TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX [3]

	10 CFR 50 App. A GDC 35	10 CFR 50 App. A GDC 37	10 CFR 50 App. A GDC 38	10 CFR 50 App. A GDC 40	10 CFR 50 App. A GDC 41	10 CFR 50 App. A GDC 43	10 CFR 50 App. A GDC 44	10 CFR 50 App. A GDC 46	10 CFR 50 App. A GDC 50	10 CFR 50 App. A GDC 54	10 CFR 50 App. A GDC 55	10 CFR 50 App. A GDC 56	10 CFR 50 App. A GDC 57
Reactor Water Cleanup System											X		
Leak Detection Systems	X									X	X	X	
Reactor Shutdown Cooling Mode (RHR)													
Spent Fuel Cooling and Cleanup System													
ARI System													
Safety Related Display Instrumentation			X		X					X	X	X	X
RHRS Containment Spray Cooling Mode			X	X					X				
Recirculation Pump Trip													
RHRS Suppression Pool Cooling Mode			X	X					X				

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TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX [3]

	IEEE 279-1971	IEEE 308-1974	IEEE 323-1971	IEEE 336-1971	IEEE 338-1971	IEEE 344-1971	IEEE 379-1972	IEEE 384-1974
RPS	X			X	X	X	X	X
PCRVICS	X	X	X	X	X	X	X	
ECCS	X	X	X		X	X	X	
NMS	X		X		X	X	X	
Reactor Manual Control System	(1)							
Reactor Vessel Instrumentation	X							
Recirculation Flow Control								
Feedwater Control System								
Process Computer	(2)							
RCIC	X	X	X		X	X		
Standby Liquid Control System	X		X		X	X		
Reactor Water Cleanup System								

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TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX [3]

	IEEE 279-1971	IEEE 308-1974	IEEE 323-1971	IEEE 336-1971	IEEE 338-1971	IEEE 344-1971	IEEE 379-1972	IEEE 384-1974
NSSS Leak Detection System	X	X	X	X	X	X	X	
Reactor Shutdown Cooling Mode (RHR)	X	X	X		X	X	X	
Spent Fuel Pool Cooling and Cleanup System								X
ARI System				X	X	X		X
Safety Related Display Instrumentation	X	X	X	X	X			
RHRS Containment Spray Cooling Mode	X	X	X	X	X	X	X	
Recirculation Pump Trip	X	4	X		X	X	X	
RHRS Suppression Pool Cooling Mode	X	X	X		X	X	X	

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TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX (3)

	RG 1.6-03/10/71	RG 1.11-03/10/71	RG 1.22-02/17/72	RG 1.29-02/76	RG 1.30-08/72	RG 1.32-08/72	RG 1.47-05/73	RG 1.53-[illegible]/73	RG 1.62-10/73	RG 1.56-06/73
RPS			X	X	X		X	X	X	
PCRVICS			X	X	X		X	X	X	
ECCS	X		X	X	X		X	X	X	
NMS			X				X	X		
Reactor Manual Control System										
Reactor Vessel Instrumentation										
Recirculation Flow Control										
Feedwater Control System										
Process Computer										
RCIC	X		X	X	X		X	X	X	
Standbv Liauid Control Svstem			X	X	X		X		X	

Table Rev. 56

TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX [3]

	RG 1.6-03/10/71	RG 1.11-03/10/71	RG 1.22-02/17/72	RG 1.29-[illegible]	RG 1.30-[illegible]8/72	RG 1.32-[illegible]8/72	RG 1.47-05/73	RG 1[illegible]3-[illegible]/73	RG 1[illegible]2-10/73	RG 1[illegible]/73
Reactor Water Cleanup System		X								X
NSSS Leak Detection Systems	X		X				X	X		
Reactor Shutdown Cooling Mode (RHR)	X		X							
Spent Fuel Cooling and Cleanup System										
ARI System			X		X			X	X	
Safety Related Display Instrumentation			X			X	X			
RHRS Containment Spray Cooling Mode	X		X							
Recirculation Pump Trip			X	X		4	X	X		
Containment Cooling System			X	X		X				
RHRS Suppression Pool Cooling Mode			X	X	X		X	X	X	

Table Rev. 56

TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX (3)

	RG 1.63-10/73	RG 1.68-11/73	RG 1.70-09/75	RG 1.75-01/75	RG 1.89-11/74	RG 1.96-05/75
RPS		X	X			
PCRVICS			X			
ECCS			X			
NMS			X			
Reactor Manual Control System			X			
Reactor Vessel Instrumentation			X			
Recirculation Flow Control			X			
Feedwater Control System			X			
Process Computer			X			
RCIC			X			
Standby Liquid Control System			X			
Reactor Water Cleanup System			X			

Table Rev. 56

TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX (3)

	RG 1.63-10/73	RG 1.68-11/73	RG 1.70-09/75	RG 1.75-01/75	RG 1.89-11/74	RG 1.96-05/75
NSSS Leak Detection Systems			X			
Reactor Shutdown Cooling Mode (RHR)			X			
Spent Fuel Cooling and Cleanup System			X			
ARI System			X	X		
Safety Related Display Instrumentation		X	X			
RHRS Containment Spray Cooling Mode			X			
Recirculation Pump Trip			X			
RHRS Suppression Pool Cooling Mode			X			

Table Rev. 56

TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX [3]

	BTPEICSB-3	BTPEICSB-10	BTPEICSB-21
RPS		X	X
PCRVICS		X	X
ECCS	X	X	X
NMS		X	
Reactor Manual Control System			
Reactor Vessel Instrumentation			
Recirculation Flow Control			
Feedwater Control Svstem			
Process Computer			
RCIC		X	X
Standby Liquid Control System		X	

Table Rev. 56

TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX [3]

	BTPEICSB 3	BTPEICSB 10	BTPEICSB-21
Reactor Water Cleanup System			
NSSS Leak Detection Systems		X	X
Reactor Shutdown Cooling System (RHR)	X	X	X
Spent Fuel Pool Cooling and Cleanup System			
ARI System		X	
Safety Related Display Instrumentation			X
RHRS Containment Spray Cooling Mode		X	X
Recirculation Pump Trip		X	X
RHRS Suppression Pool Cooling Mode		X	

Table Rev. 56

TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX (3)

	BTPEICSB-22	BTPEICSB-23	BTPEICSB-26
RPS	X		X
PCRVICS	X		
ECCS	X		
NMS	X		
Reactor Manual Control System			
Reactor Vessel Instrumentation			
Recirculation Flow Control			
Feedwater Control Svstem			
Process Computer			
RCIC	X		
Standby Liquid Control System	X		
Reactor Water Cleanup System			

Table Rev. 56

TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX (3)

	3T@EICSB-22	BTPEICSB-23	3T@EICSB-26
NSSS Leak Detection Systems	X		
Reactor Shutdown Cooling System (RHR)	X		
Spent Fuel Pool Cooling and Cleanup System			
ARI Svstem	X		
Safety Related Display Instrumentation		X	
RHRS Containment Spray Cooling Mode	X		
Recirculation Pump Trip	X		
RHRS Suppression Pool Cooling Mode	X		

Table Rev. 56

TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX (3)

NOTES:

1. Interlock functions for Rod Withdrawal Block (RBM) are required to meet specific NRC requirements, rather than IEEE-279.

2. Process Computer contains Rod Worth Minimizer Program, which is a portion of the Rod Withdrawal Block Interlock function.

3. This table indicates applicability of codes and standards to the systems. The degree of conformance is stated in the conformance section for each system.

4. Compliance to IEEE 308-1974 and Regulatory Guide 1.32-1972 does not apply to the logic system, which is fail safe. Class 1E DC Control Power is provided to energize the breaker trip coils.

TABLE 7.1-4

REACTOR PROTECTION SYSTEM CODES AND STANDARDS

	IEEE-279-1971	IEEE-336-1971	IEEE-338-1971	IEEE-344-1971 (1)	IEEE-379-1972	IEEE-384-1974	R.G. 1.89 (1)	R. G. 1.22	R. G. 1.29	R. G. 1.47	R. G. 1.53	R. G. 1.62	R. G. 1.75
SCRAM DISCHARGE VOLUME	X		X	X	X	X							
MSL ISOLATION VALVE CLOSURE	X		X	X	X	X				X	X		
TURBINE STOP VALVE CLOSURE	X		X	X	X	X					X		
TURBINE CONTROL VALVE FAST CLOSURE	X		X	X	X	X					X		
REACTOR LOW WATER LEVEL	X		X	X	X	X							
NEUTRON MONITORING SYSTEM IRM	X		X	X	X	X							
NEUTRON MONITORING SYSTEM APRM	X		X	X	X	X							
DRYWELL HIGH PRESSURE	X		X	X	X	X							
REACTOR HIGH PRESSURE	X		X	X	X	X							
MANUAL SWITCH INPUTS	X		X	X	X	X							
BYPASS INPUTS	X		X	X	X	X							
TRIP LOGIC TRIP ACTUATOR OUTPUTS	X		X	X	X	X		X					

(1) See Table 7.1-3 notes for further discussion

TABLE 7.1-4

REACTOR PROTECTION SYSTEM CODES AND STANDARDS

	GDC 13	GDC 19	GDC 20	GDC 21	GDC 22	GDC 23	GDC 24	GDC 25	GDC 29	R. G. 1.30	R. G. 1.68
SCRAM DISCHARGE VOLUME	X		X	X	X	X	X	X	X	X	X
MSL ISOLATION VALVE CLOSURE	X		X	X	X	X	X		X	X	X
TURBINE STOP VALVE CLOSURE	X		X	X	X	X	X		X	X	X
TURBINE CONTROL VALVE FAST CLOSURE	X		X	X	X	X	X		X	X	X
REACTOR LOW WATER LEVEL	X		X	X	X	X	X		X	X	X
NEUTRON MONITORING SYSTEM IRM	X		X	X	X	X	X	X	X	X	X
NEUTRON MONITORING SYSTEM APRM	X		X	X	X	X	X	X	X	X	X
DRYWELL HIGH PRESSURE	X		X	X	X	X	X		X	X	X
REACTOR HIGH PRESSURE	X		X	X	X	X	X		X	X	X
MANUAL SWITCH INPUTS		X		X	X	X	X	X	X		
BYPASS INPUTS			X	X	X	X	X	X	X		
TRIP LOGIC											
TRIP ACTUATOR OUTPUTS			X	X	X	X	X	X	X		

TABLE 7.1-5

PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM CODES AND STANDARDS

	10 CFR 50.34	10 CFR 50.36	10 CFR 50.55A	GDC 1	GDC 2	GDC 3	GDC 4	GDC 5	GDC 10	GDC 13	GDC 19	GDC 20	GDC 21	GDC 22
REACTOR LOW WATER LEVEL	X	X	X	X	X	X	X	X	X	X		X	X	X
MSL HIGH RADIATION	X	X	X	X	X	X	X	X		X		X	X	X
MSL HIGH FLOW	X	X	X	X	X	X	X	X		X		X	X	X
MSL TUNNEL HIGH TEMPERATURE	X	X	X	X	X	X	X	X		X		X	X	X
MSL TUNNEL HIGH DIFF. TEMPERATURE	X	X	X	X	X	X	X	X		X		X	X	X
REACTOR LOW PRESSURE	X	X	X	X	X	X	X	X	X	X		X	X	X
DRYWELL HIGH PRESSURE	X	X	X	X	X	X	X	X		X		X	X	X
CONTAINMENT EXHAUST VENT PLENUM MON.	X	X	X	X	X	X	X	X		X		X	X	X
REACTOR WATER CLEANUP LOOP HIGH DIFF. FLOW	X	X	X	X	X	X	X	X		X		X	X	X
REACTOR WATER CLEANUP LOOP HIGH SPACE TEMPERATURE	X	X	X	X	X	X	X	X		X		X	X	X

(1) Refer to notes on Table 7.1-3

TABLE 7.1-5

PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM CODES AND STANDARDS

	GDC 23	GDC 24	GDC 29	GDC 35	GDC 54	GDC 55	GDC 56	IEEE-279-1971	IEEE 323-1971	IEEE-336-1971	IEEE-338-1971	IEEE-344-1971
REACTOR LOW WATER LEVEL	X	X	X			X	X	X	X	X	X	X
MSL HIGH RADIATION	X	X	X			X	X	X	X	X	X	X
MSL HIGH FLOW	X	X	X		X	X	X	X	X	X	X	X
MSL TUNNEL HIGH TEMPERATURE	X	X	X		X	X	X	X	X	X	X	X
MSL TUNNEL HIGH DIFF. TEMPERATURE	X	X	X		X	X	X	X	X	X	X	X
REACTOR LOW PRESSURE	X	X	X			X	X	X	X	X	X	X
DRYWELL HIGH PRESSURE	X	X	X	X		X	X	X	X	X	X	X
CONTAINMENT EXHAUST VENT PLENUM MON.	X	X	X					X	X	X	X	X
REACTOR WATER CLEANUP LOOP HIGH DIFF. FLOW	X	X	X		X	X	X	X	X	X	X	X
REACTOR WATER CLEANUP LOOP HIGH SPACE TEMPERATURE	X	X	X		X	X	X	X	X	X	X	X

TABLE 7.1-5

PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM CODES AND STANDARDS

	IEEE-379-1972	R. G. 1.22	R. G. 1.29	R. G. 1.30	R. G. 1.47	R. G. 1.53	R. G. 1.62	R. G. 1.70	BTPEICSB10	BTPEICSB21
REACTOR LOW WATER LEVEL	X	X	X	X		X		X	X	X
MSL HIGH RADIATION	X	X	X	X		X		X	X	X
MSL HIGH FLOW	X	X	X	X		X		X	X	X
MSL TUNNEL HIGH TEMPERATURE	X	X	X	X		X		X	X	X
MSL TUNNEL HIGH DIFF. TEMPERATURE	X	X	X	X		X		X	X	X
REACTOR LOW PRESSURE	X	X	X	X		X		X	X	X
DRYWELL HIGH PRESSURE	X	X	X	X		X		X	X	X
CONTAINMENT EXHAUST VENT PLENUM MON.	X	X	X	X		X		X	X	X
REACTOR WATER CLEANUP LOOP HIGH DIFF. FLOW	X	X	X	X		X		X	X	X
REACTOR WATER CLEANUP LOOP HIGH SPACE TEMPERATURE	X	X	X	X		X		X	X	X

TABLE 7.1-5

PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM CODES AND STANDARDS

	10 CFR 50.34	10 CFR 50.36	10 CFR 50.55a	GDC 1	GDC 2	GDC 3	GDC 4	GDC 5	GDC 10	GDC 13	GDC 19	GDC 20	GDC 21	GDC 22
REACTOR WATER CLEANUP LOOP HIGH SPACE DIFF. TEMPERATURE	X	X	X	X	X	X	X	X		X		X	X	X
REACTOR WATER CLEANUP FILTER DEMIN. INLET HIGH TEMPERATURE														
RHR HIGH FLOW	X	X	X	X	X	X	X	X		X		X	X	X
CONDENSER LOW VACUUM	X	X	X	X	X	X	X	X		X		X	X	X
MANUAL SWITCH INPUTS	X	X	X	X	X	X	X	X			X		X	X
BYPASS INPUTS	X	X	X	X	X	X	X	X	X	X		X	X	X
TRIP LOGIC TRIP ACTUATOR OUTPUTS	X	X	X	X	X	X	X	X	X	X		X	X	X

TABLE 7.1-5

PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM CODES AND STANDARDS

	BTPISB22
REACTOR LOW WATER LEVEL	X
MSL HIGH.RADIATION	X
MSL HIGH FLOW	X
MSL TUNNEL HIGH TEMPERATURE	X
MSL TUNNEL HIGH DIFF. TEMPERATURE	X
REACTOR LOW PRESSURE	X
DRYWELL HIGH PRESSURE	X
CONTAINMENT EXHAUST VENT PLENUM MON.	X
REACTOR WATER CLEANUP LOOP HIGH DIFF. FLOW	X
REACTOR WATER CLEANUP LOOP HIGH SPACE TEMPERATURE	X

TABLE 7.1-5

PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM CODES AND STANDARDS

	GDC 23	GDC 24	GDC 29	GDC 35	GDC 54	GDC 55	GDC 56	IEEE-279-1971	IEEE-323-1971	IEEE-336-1971	IEEE-338-1971	IEEE-344-1971
REACTOR WATER CLEANUP LOOP HIGH SPACE DIFF. TEMPERATURE	X	X	X		X	X	X	X	X	X	X	X
REACTOR WATER CLEANUP FILTER DEMIN. INLET HIGH TEMPERATURE												
RHR HIGH FLOW	X	X	X		X	X	X	X	X	X	X	X
CONDENSER LOW VACUUM	X	X	X			X	X	X	X	X	X	X
MANUAL SWITCH INPUTS	X	X	X					X	X	X	X	X
BYPASS INPUTS	X	X	X		X	X	X	X	X	X	X	X
TRIP LOGIC TRIP ACTUATOR OUTPUTS	X	X	X		X	X	X	X	X	X	X	X

TABLE 7.1-5

PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM CODES AND STANDARDS

	IEEE-379-1972	R. G. 1.22	R. G.1.29	R. G. 1.30	R. G. 1.47	R. G. 1.53	R. G. 1.62	R. G. 1.70	BTPEICSB10	BTPEICSB21
REACTOR WATER CLEANUP LOOP HIGH SPACE DIFF. TEMPERATURE	X	X	X	X		X		X	X	X
REACTOR WATER CLEANUP FILTER DEMIN. INLET HIGH TEMPERATURE										
RHR HIGH FLOW	X	X	X	X		X		X	X	X
CONDENSER LOW VACUUM	X	X	X	X		X		X	X	X
MANUAL SWITCH INPUTS	X	X	X	X		X	X	X	X	X
BYPASS INPUTS	X	X	X	X		X		X	X	X
TRIP LOGIC TRIP ACTUATOR OUTPUTS	X	X	X	X		X		X	X	X

TABLE 7.1-5

PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM CODES AND STANDARDS

	BTPEICSB22
REACTOR WATER CLEANUP LOOP HIGH SPACE DIFF. TEMPERATURE	X
REACTOR WATER CLEANUP FILTER DEMIN. INLET HIGH TEMPERATURE	X
RHR HIGH FLOW	X
CONDENSER LOW VACUUM	X
MANUAL SWITCH INPUTS	X
BYPASS INPUTS	X
TRIP LOGIC TRIP ACTUATOR OUTPUTS	X

TABLE 7.1-6

HIGH PRESSURE ECCS (HPCI, ADS A, ADS B NETWORK)
CODES AND STANDARDS

	REACTOR LOW WATER LEVEL	PRIMARY CONTAINMENT HIGH PRESSURE	HPCI EMERG BUS VOLTAGE	HPCI FLOW SUFFICIENT	HPCI BATTERY VOLTAGE	ADS A BATTERY VOLTAGE	ADS A AC INTLK PERMISSIVE	ADS A TIMER	ADS B BATTERY VOLTAGE	ADS B AC INTLK PERMISSIVE	ADS B TIMER	MANUAL SWITCH INPUTS	BYPASS INPUTS	TRIP LOGIC, TRIP ACTUATOR OUTPUTS
IEEE-279-1971	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
IEEE-308-1974			XN		XN	XN			XN					
IEEE-323-1971	X	X	X	X	XN	XN	X	X	XN	X	X	X	X	X
IEEE-336-1971	X	X	X	X	XN	XN	X	X	XN	X	X	X	X	X
IEEE-344-1971	X	X	X	X	XN	XN	X	X	XN	X	X	X	X	X
IEEE-379-1972	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
RG 1.6			XN		XN	XN			XN					
RG 1.22	X	X	X	X	X	X	X	X	X	X	X	X		
RG 1.29														
RG 1.32														
RG 1.47														
RG 1.53	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
RG 1.62														
GDC 13	X	X	X	X	X	X	X		X	X				
GDC 19			X									X		
GDC 20	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN		XN	
GDC 21	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 22	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 23	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 24	X	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 29	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 34	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 35	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 37	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN

X - APPLICABLE
XN - APPLICABLE ON A NETWORK BASIS

TABLE 7.1-7

LOW PRESSURE ECCS (CS, RHR NETWORK)
CODES AND STANDARDS

	REACTOR LOW WATER LEVEL	DRYWELL HIGH PRESSURE	HPCI EMERGENCY BUS VOLTAGE	HPCI FLOW SUFFICIENT	HPCI BATTERY VOLTAGE	CS/RHR A BATTERY VOLTAGE	CS/RHR A EMERG BUS VOLTAGE	CS/RHR A FLOW SUFFICIENT	CS/RHR A INJECTION VALVE P	RHR B BATTERY VOLTAGE	RHR B BUS VOLTAGE	RHR B SUFFICIENT	RHR B INJEC-TION VALVE P	MANUAL SWITCH	BYPASS INPUTS	TRIP LOGIC TRIP ACTUATOR INPUTS
IEEE-279-1971	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
IEEE-308-1971			XN		XN	XN	XN			XN	XN					
IEEE-323-1971	X	X	X	X	XN	XN	XN	X	X	XN	XN	X	X	X	X	X
IEEE-336-1971	X	X	X	X	XN	XN	XN	X	X	XN	XN	X	X	X	X	X
IEEE-344-1971	X	X	X	X	XN	XN	XN	X	X	XN	XN	X	X	X	X	X
IEEE-379-1972	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
RG 1.6			XN		XN	XN	XN			XN	XN					
RG 1.22	X	X	X	X	X	X	X	X	X	X	X	X	X	X		X
RG 1.29																
RG 1.32																
RG 1.47																
RG 1.53																
RG 1.62																
GDC 13	X	X	X	X	X	X	XN	X	X	X	XN	X	X			
GDC 19			X											X		
GDC 20	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN		XN	XN
GDC 21	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 22	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 23	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 24	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 29	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 34	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 35	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 37	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN

X = APPLICABLE
XN = APPLICABLE ON NETWORK BASIS

TABLE 7.1-8

PROCESS RADIATION MONITORING
CODES AND STANDARDS

SYSTEM	CODE OR STANDARD
Main Steamline	10 CFR 50.34
	10 CFR 50.36
	10 CFR 50.55a
	IEEE-279-1971
	IEEE-323-1971(1)
	IEEE-338-1975
	IEEE-344-1971(1)
	IEEE-379-1972
	R.G. 1.22
	R.G. 1.29
	R.G. 1.30
	R.G. 1.47
	R.D. 1.53
	GDC 1
	GDC 2
	GDC 3
	GDC 4
	GDC 13
	GDC 20
	GDC 21
	GDC 22
	GDC 23
	GDC 24
	GDC 29
	BTPEICSB-10
	BTPEICSB-21
	BTPEICSB-22

See Table 7.1-3 for Notes

TABLE 7.1-9

LEAK DETETION SYSTEM CODES AND STANDARDS

	Systems Affected	10 CFR 50.55a	10 CFR 50.34	10 CFR 50.36	IEEE-279-1971	IEEE-323-1971 (1)	IEEE-338-1971	IEEE-344-1971 (1)	IEEE-379-1972	RG 1.22	RG 1.29
HIGH TEMPERATURE AND Δ TEMPERATURE	MSL										
	RCIC	X	X	X	X	X	X	X	X	X	X
	RHR										
	RWCU										
	HPCI										
HIGH FLOW [3]	MSL	X	X	X	X	X	X	X	X	X	X
	RCIC										
	RHR										
	RWCU										
	HPCI										
LOW RV WATER LEVEL [3]	MSL	X	X	X	X	X	X	X	X	X	X
HIGH PRESSURE [3]	RCIC	X	X	X	X	X	X	X	X	X	X
	RHR										
HIGH Δ FLOW	RWCU	X	X	X	X	X	X	X	X	X	X
SUMP FILL RATE [3]		X	X	X			X				
RECIRCULATION PUMP LEAK PRESSURE FLOW [3]		X	X	X			X				
SAFETY RELIEF VALVE DISCHARGE TEMPERATURE		X	X	X			X				

TABLE 7.1-9

LEAK DETETION SYSTEM CODES AND STANDARDS

	Systems Affected	RG 1.47	RG 1.53	RG 1.68	RG 1.75	GDC 1	GDC 2	GDC 3	GDC 4	GDC 10	GDC 13	GDC 19	GDC 20	GDC 21	GDC 22
HIGH TEMPERATURE AND Δ TEMPERATURE	MSL														
	RCIC	X	X	X	X	X	X	X	X	X	X	X	X	X	X
	RHR														
	RWCU														
	HPCI														
HIGH FLOW [3]	MSL	X	X	X	X	X	X	X	X	X	X	X	X	X	X
	RCIC														
	RHR														
	RWCU														
	HPCI														
LOW RV WATER LEVEL [3]	MSL	X	X	X		X	X	X	X	X	X	X	X	X	X
HIGH PRESSURE [3]	RCIC	X	X	X	X	X	X	X	X	X	X	X	X	X	X
	RHR														
HIGH Δ FLOW	RWCU	X	X	X	X	X	X	X	X	X	X	X	X	X	X
SUMP FILL RATE [3]						X	X	X	X		X				
RECIRCULATION PUMP LEAK PRESSURE FLOW [3]						X	X	X	X	X	X				
SAFETY RELIEF VALVE DISCHARGE TEMPERATURE						X	X	X	X	X	X				

TABLE 7.1-9

LEAK DETETION SYSTEM CODES AND STANDARDS

	Systems Affected	GDC 23	GDC 24	GDC 29	GDC 30	GDC 33	GDC 34	GDC 35	GDC 54	GDC 55	GDC 56
HIGH TEMPERATURE AND Δ TEMPERATURE	MSL										
	RCIC	X	X	X	X	X	X		X	X	X
	RHR										
	RWCU										
	HPCI										
HIGH FLOW (3)	MSL	X	X	X	X	X	X		X	X	
	RCIC										
	RHR										
	RWCU										
	HPCI										
LOW RV WATER LEVEL (3)	MSL	X	X	X	X	X	X		X	X	
HIGH PRESSURE (3)	RCIC	X	X	X	X	X	X		X	X	X
	RHR										
HIGH Δ FLOW	RWCU	X	X	X	X	X	X		X	X	
SUMP FILL RATE (3)					X	X		X			
RECIRCULATION PUMP LEAK PRESSURE FLOW (3)					X	(2)					
SAFETY RELIEF VALVE DISCHARGE TEMPERATURE					X	X					

(1) See Table 7.1-3 for further discussion.

(2) Flow only.

(3) Contribute to drywell leak detection.

TABLE 7.1-10

REACTOR PROTECTION SYSTEM
SENSOR SUFFIX LETTERS AND DIVISION ALLOCATION*

TOTAL NUMBER SENSORS	DIVISION 1A	DIVISION 1B	DIVISION 2A	DIVISION 2B
	Trip Logic A1	Trip Logic B1	Trip Logic A2	Trip Logic B2
4	A	B	C	D
8	A,E	B,F	C,G	D,H
16	A,E,J,N	B,F,K,P	C,G,L,R	D,H,M,S
	Part of Trip System A	Part of Trip System B	Part of Trip System A	Part of Trip System B

* This Division does not apply to the APRM system which must have a special four group arrangement to allow for maintenance bypassing without violating the single failure criteria (See Table 7.1-11).

TABLE 7.1-11

FOUR DIVISION GROUPING FOR NEUTRON MONITORING SYSTEM

STANDARD 4 PENET GROUPING	A		C		D		B	
WIREWAY	NA		NB		NC		ND	
SRM	A		B		C		D	
IRM	A	E	B	F	C	G	D	H
APRM CHANNEL DESIG.	1		2		3		4	
RPS TRIP LOGIC	A1		B1		A2		B2	

NOTES:

1. Penetrations across top of table serve APRM's, IRM'S, SRM's and the RPS Trip Logics directly below them.
2. Horizontal zoning represents LPRM Cable distribution to APRM's from various penetrations, e.g., Penetration B carries cable for LPRM's going to APRM Channel 4.

TABLE 7.1-12

EMERGENCY CORE COOLING SYSTEM AND RCIC

SENSOR SUFFIX LETTERS AND DIVISION ALLOCATION

ENERGIZE-TO-OPERATE

DIVISION I		DIVISION II	
SENSOR SUFFIX LETTERS		SENSOR SUFFIX LETTERS	
A,	C	B,	D
Operate ECCS Division 1 directly and ECCS Division 2 through isolation devices		Operate ECCS Division 2 directly and ECCS Division 1 and RCIC through isolation devices	