7.5 SAFETY-RELATED DISPLAY INSTRUMENTATION (SRDI)

Safety-Related Display Instrumentation (SRDI) required for safe functioning of the plant during normal operating and accident conditions is incorporated in a control room complex named the Advanced Control Room (ACR). It is necessary to consider the ACR as a whole to verify conformance to the requirements.

The ACR is a complex of major components, provided with the NSSS, for monitoring and controlling two units and providing safety functions. The entire complex consists of the Power Generation Control Complex (PGCC) in the upper relay room at the 754/ft. 0/in. level and the lower relay room at the 698/ft. 0/in. level, the Plant Integrated Computer System at the 698/ft. 0/in. level and the plant-operator interface at the 729/ft. 0/in. level.

a) PGCC

The PGCC provides support and interconnections to the systems panels of the upper and lower relay rooms and the computer room of the ACR complex. The plant-operator interface noted below is not mounted on PGCC, however, all other principles of the PGCC concept, e.g., separation, are used.

b) Plant-Operator Interface

Major components are in the main control room arranged as shown on Dwg. A-105, Sh. 1 and described on Figure 7.7-13A and include the following:

Unit Operating Benchboard Panel (C651/H12-P680) - houses controls, hardwired displays, the control rod position display and process displays which are computer generated from the Plant Integrated Computer System described briefly below and more completely in Section 7.7. The combination of displays on this panel and panel (C652/H12-P678), Standby Information Panel, are arranged by system and are used for start up, normal operation, and shut down.

See Section 7.7 for a description of the Plant Integrated Computer System, in regard to the displays on all control room panels.

Standby Information Panel (C652/H12-P678) - houses hardwired indicators and recorders required to start up, run, and shut down the plant. It is a hardwired backup to the Plant Integrated Computer System.

Reactor Core Cooling System Benchboard (C601/H12-P601) - houses hardwired indicators, recorders, manual controls, and annunciators for ESF systems including containment atmosphere systems.

Unit Services Benchboard (C668/H12-P870) - houses hardwired indicators, recorders, annunciators and controls for unit BOP system's functions which do not require the operator's immediate attention during normal operation of the power plant. Functions on this panel have been determined to be long time response functions.

Plant Operating (Common Plant) Benchboard (C653/H12-P853) - houses hardwired indicators, recorders, controls and annunciators for systems common to both units. Manual controls for the diesel generators are located here. It also houses two displays connected to the Plant Integrated Computer System.

Unit Monitoring Console (C684/C92-P628) - provides the unit operator with sit down surveillance of the unit operating benchboard and access to the Plant Integrated Computer System.

Safety Parameter Display System/Plant Monitoring Console (C667) - provides sit down surveillance of both units and access to Plant Computer System displays and Plant Computer Functions as well as SPDS displays.

Panels which support the primary plant-operator interface are mounted in back rows of the main control room and on PGCC floor modules on floors above and below the main control room.

The annunciator system is a hardwired system which provides the operator with the alarm information required for unit operation, startup, and shutdown. This system is independent of the Plant Integrated Computer System, is not part of the SRDI, and is not Class 1E.

c) Plant Integrated Computer System

The Plant Integrated Computer System is primarily used for monitoring unit operation by generating graphic displays to display optimized information for operator surveillance. In addition, it is capable of generating other displays, making NSSS and BOP calculations, recording historical data, logging data, and performing off-line functions.

In addition to the NSSS calculational capabilities, a Reactor Data Analysis System (RDAS) is available to monitor core conditions. The RDAS will receive data from the Plant Integrated Computer System and will be capable of providing information to be displayed in the ACR. The Plant Integrated Computer System and RDAS are not part of the SRDI and are not Class 1E.

The Plant Integrated Computer System makes use of redundant computers for critical applications to maximize availability. As stated above, the information important to safety displayed by the Plant Integrated Computer System, is also displayed on hardwired indicators, recorders and alarm annunciators. Alarms in Plant Integrated Computer System displays are redundant to those provided by the hardwire annunciator system. While the operator makes primary use of the computer generated displays during normal operation, he is not dependent on computers or annunciators to operate the plant safely during startup, steady-state operation, or any design basis events.

Plant Integrated Computer System displays are designed to be used on the ACR benchboard panels, and are organized by system and plant mode of operation. The operator can select displays on most benchboard displays, appropriate to each benchboard area and plant mode of operation, with one select operation from a control position on each benchboard associated with that display. The operator can individually call up any available display on any display for the Plant Integrated Computer System.

7.5.1a Description

This section describes the hardwired displays noted above in the ACR description. It does not describe computer generated displays. SRDI is described on the basis of NSSS and non-NSSS responsibility by system. NSSS and non-NSSS share panels and use the same types of instruments which were provided with the NSSS. (Some of the instrumentation described below may be non-SRDI, and is provided to give a more complete picture of the totality of information available to the Operator in the ACR. Reference should be made to Tables 7.5-1 through 7.5-7a to determine which instrumentation is part of the SRDI.)

Therefore, control room panels and instruments, unless otherwise noted, use NSSS qualification and separation techniques.

As noted in Section 7.1, sections with an "a" describe NSSS and "b" denote non-NSSS.

7.5.1a.1 General Description of NSSS Safety-Related Display Instrumentation

This section describes the instrumentation which provides information to the operator to enable him to perform required safety functions.

The Safety-Related Display Instrumentation is listed in Table 7.5-1. It tabulates SRDI equipment which may be illustrated on the various system figures located in Sections 7.2, 7.3, 7.4, and 7.6.

The instrumentation and ranges shown in Tables 7.5-1 through 7.5-7a are selected on the basis of giving the reactor operator information to perform manual safety functions and yet the capability to track process variables pertinent to safety during expected operational perturbations.

Separation of redundant display instrumentation and electrical isolation of redundant sensors and channels is illustrated in the Elementary Diagrams, Electrical Schematics and the Loop Diagrams associated with the individual circuits. Additional information on the redundancy of monitored variables and component sensors and channels can be found in Design Specifications, P&ID and Design Basis Documents.

7.5.1a.2 Normal Operation

The information channel ranges and hardwired indicators and recorders were selected on the basis of giving the reactor operator the necessary information to perform all the normal plant startup, steady-state maneuvers, with the required precision and to be able to track all the process variables pertinent to safety during expected operational perturbations.

7.5.1a.3 Abnormal Transient Occurrences

The ranges of indicators and recorders provide adequate information for all abnormal transient events.

7.5.1a.4 Accident Conditions

The DBA-LOCA is the most extreme postulated operational action event. Adequate instrumentation is provided from the standpoint of operator action, information, and event tracking requirements, in order to cover all other design-basis events or incident requirements.

7.5.1a.4.1 Initial Accident Event

The design basis of all ESF to mitigate the accident event condition takes into consideration that no operator action or assistance is assumed for the first twenty minutes of the event with one exception. 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. This requirement, therefore, makes it mandatory that all protective action necessary in the first twenty minutes be "automatic." Therefore, although continuous tracking of process variables is available, no operator action based on them is required.

7.5.1a.4.2 Post-Accident Tracking

No operator action (and, therefore, post-accident information) is required for at least twenty minutes following an accident although the various monitoring devices are continuously tracking and indicating important parameter information and displaying it to the operator as well as recording appropriate data.

The DBA-LOCA serves as the envelope accident sequence event to provide and demonstrate the plant's post-accident tracking capabilities. All other accidents have less severe and limiting tracking requirements. Refer to Chapter 15 "Accident Analysis."

The following process instrumentation provides information to the operator after a DBA-LOCA to monitor reactor parameters.

7.5.1a.4.2.1 Reactor Water Level

(1) Two independent wide range water level signals are transmitted to and recorded on two separate recorders. The differential pressure transmitters have one side connected to a condensing chamber reference leg and the other side connected directly to a vessel nozzle for the variable leg. The water level system is uncompensated for variation in reactor water density and is calibrated to be most accurate at operational pressure and temperature conditions at which it is used. The range of the recorded level is from the top of the feedwater control range (just above the high level turbine trip point) down to a point near the top of the active fuel. The power sources for the two channels are separate, divisionalized Class 1E buses.

(2) Two fuel zone water level signals are transmitted from two independent differential pressure transmitters. One signal goes to a fuel zone water level indicator and the other water level signal goes to a fuel zone water level recorder. The differential pressure transmitters have one side connected to a condensing chamber reference leg and the other side connected directly to the bottom tap of a calibrated jet pump for the variable leg. The fuel zone water level system is uncompensated for variation in reactor water density and is calibrated to be most accurate at saturated atmospheric conditions at which it is used. The level range is from over the top of the active fuel to near the bottom of the active fuel. The ranges of the wide range level and the fuel zone level will overlap. Power sources are as stated in the previous paragraph.

(3) Two extended range water level signals are transmitted from two independent differential pressure transmitters. Each level transmitter is connected to a dual indicator which displays both wide and extended range. The differential transmitter reference leg is tubed to a condensing chamber and the variable leg to vessel nozzle. The water level system is

uncompensated for variation in reactor water density and is calibrated to be most accurate at operational pressure and temperature conditions at which it is used. The level range is from over the top of the steam lines to near the top of the active fuel. The range of the extended level will overlap the wide range level. Power sources are as stated in the first paragraph.

(4) Non-condensible gases may exist in the reference legs of the nuclear boiler instrumentation system. During a rapid or normal depressurization of the reactor, these gases may come out of solution displacing water from the reference legs and leading to false high level indications. The non-condensible gases are believed to be H_2 and O_2 generated by the dissociation of H_2O from neutron flux. (Reference NRC Generic Letter 92-04 and Information Notice 93-27.) All three of the following conditions must exist in order for these level indication errors to occur:

1. An elevated level of non-condensible gases in the condensing chamber,

2. A migration of the gases into the reference leg, and

3. A Reactor Pressure Vessel depressurization (below 450 psig).

Non-condensible gases will exist in the condensing chamber during normal operations. Vessel depressurizations will inevitably occur during the life of the plant and migration of gases may be undetectable. To prevent level indication errors from occurring, the concentration of non-condensibles in the reference legs must be kept below 150 ppmv. This value has been determined to be acceptable by Boiling Water Reactor Owner's Group (BWROG) testing.

Vent lines from condensing chambers XY-B21-1/2D004A and XY-B21-1/2D004B to the variable leg piping from vessel nozzles N11A and N11B and from condensing chambers XY-B21-1/2D002 and XY-1/24202 to the RPV head vent line are installed to prevent non-condensible gas build-up in the reference leg by providing a path to sweep away any gases which may build-up in the chamber. This eliminates condition 1 above, by assuming that the non-condensible gases never build-up in the steam space of the condensing chamber.

7.5.1a.4.2.2 Reactor Pressure

Three independent reactor pressure signals (ranged 0 to 1500 psig) are transmitted and indicated on control room panels. Two signals are recorded on two separate recorders. Power sources are as stated in Subsection 7.5.1a.4.2.1.

7.5.1a.4.2.3 Reactor Shutdown, Isolation, and Core Cooling Indication

7.5.1a.4.2.3.1 Reactor Operator Information and Observations

The information furnished to the main control room operator permits him to assess reactor shutdown, isolation, and availability of emergency core cooling following the postulated accident. Some of the information listed below is provided by non-SRDI equipment.

(1) Operator verification that reactor shutdown has occurred may be made by observing the following indications:

a. Control rod status lamps indicating each rod fully inserted. The power source is one of the instrument AC buses.

b. Control rod scram pilot valve indicating lamps which are illuminated when the control rod scram pilot valves are energized. The power source is an RPS MG Set.

c. Neutron monitoring power range channels and recorders downscale. The power sources are RPS MG sets.

d. Annunciators for reactor protection system variables and trip logic in the tripped state. The power source is dc from a plant battery.

(2) The reactor operator may verify reactor isolation by observing the following indications:

a. Isolation valve position lamps indicating valve closure. The power source is the same as for the associated valve motor-operator.

b. Main steamline flow indication downscale. The power source is instrument AC.

c. Annunciators for the containment and reactor vessel isolation system variables and trip logic in the tripped state. The power source is DC from one or more plant batteries.

(3) Operation of the emergency core cooling and the RCIC system following the accident may be verified by observing the following indications:

a. Annunciators for HPCI, CS, RHR, ADS, RCIC sensor initiation logic trips. The power source is DC from a plant battery.

b. Flow and pressure indications for each emergency core cooling system. The power sources are independent and from the same standby buses as the driven equipment.

c. RCIC isolation valve position lights indicating to open valves. The power source is from the same bus as the valve motive power.

d. Injection valve position lights indicating either open or closed valves. The power source is the same as the valve motor.

e. Relief valve initiation circuit status by open or closed indicator lamps. The power source is the same as for the pilot solenoid.

f. Relief valve position may be inferred from reactor pressure indications. The power source is instrument AC from the standby AC systems.

g. Relief valve discharge pipe temperature monitors. The power source is instrument AC.

h. Relief valve position is indicated by acoustic monitors (see Subsection 18.1.24.3).

7.5.1a.4.2.3.2 System Operation Information-Display Equipment

(1) RCIC

Two meters, one displaying RCIC discharge flow rate and one displaying RCIC pump discharge pressure, are located in the main control room.

(2) HPCI

Three meters, one displaying HPCI discharge flow rate and one displaying HPCI pump discharge pressure, and one displaying HPCI turbine steam pressure, are located in the main control room.

(3) CS System

Two meters displaying CS flow rate are located in the main control room.

(4) RHR

One meter displaying RHR flow rate for each of the two RHR loops, and one meter displaying RHR service water flow rate for each of the two service water loops are located in the main control room.

(5) Miscellaneous

In addition to the above displays, the following also provide information to enable the reactor operator in the main control room to perform post-accident safety functions:

a. Control rod status lamps

b. Scram pilot valve status lamps

c. Neutron flux level meters.

7.5.1a.4.2.3.3 System Operation Information-Display Equipment Qualification

Environmental Qualification of Safety-Related Display Instrumentation is addressed in the Susquehanna SES Environmental Qualification Program for Class 1E Equipment. Additionally, some of the safety-related display instrumentation help to satisfy our commitment to Regulatory Guide 1.97, Rev. 2, and meet the additional requirements as committed to for that regulatory guide.

7.5.1a.4.2.4 Containment Indications

Refer to Table 7.5-3.

7.5.1b Description of non-NSSS Safety-Related Displays

Non-NSSS safety-related displays provide the operator with information to monitor certain containment conditions, non-NSSS ESF systems, and auxiliary support systems.

Description is provided in the form of Tables 7.5-2 through 7.5-7, which list details of the displays by ESF systems as previously listed.

7.5.1b.1 Containment Isolation

Refer to Table 6.2-12 for list of isolation valves. All containment isolation valves have status indication lights on control room panels (C651/H12-P680) or (C601/H12-P601).

7.5.1b.2 Combustible Gas Control

Refer to Table 7.5-3 for list of displays for containment atmosphere monitoring.

7.5.1b.3 Primary Containment Vacuum Relief

Status indication only is provided for test operation described in Subsection 7.3.1.1b.3. Containment pressures are noted in Table 7.5-3.

7.5.1b.4 Standby Gas Treatment, RX Building Recirculation and Isolation System

These systems are listed in Table 7.5-4.

7.5.1b.5 Habitability Systems

These systems are listed in Table 7.5-2.

7.5.1b.6 Auxiliary Support Systems

Emergency service water system displays are listed in Table 7.5-5. RHR service water system displays are listed in Table 7.5-6. Containment instrument gas system displays are listed in Table 7.5-7.

7.5.1b.7 Bypass Indication System

This system is established and used during normal reactor operations to control planned actions whose manual initiation would effectively disable any safety function.

The design complies with Regulatory Guide 1.47 (May 1973).

7.5.1b.7.1 System Description

The primary control method is administrative control which is exercised by the unit control room operator; however, these administrative controls are supplemented by an automated Bypass Indication System (BIS). Restricted access to various in-plant areas is also used to supplement the administrative control.
The BIS indicators annunciate on the Reactor Core Cooling System benchboard in the control room, automatically, at the system level, indicates the bypass or deliberately induced inoperability of a safety-related system.

The BIS is provided with the capability for manual initiation of each system-level indicator. This manual-entry method is used to cover system components that have not been provided with automatic BIS input capability.

The Bypass Indication System for non-NSSS Systems consists of the following:

a) Two indicator lamp boxes each consisting of 4x6 array of lights and located in the control room on the Reactor Core Cooling System benchboard. Each window, provided with dual lamps and an integral pushbutton for lamp test, will indicate a system-level bypass.

b) Two annunciator windows, located above the lamp box assemblies will alert the operator that a system-level bypass has occurred.

c) The indication of the bypass status of components, systems, channels, and/or divisions is provided on a backrow panel in the main control room. This panel contains the hardware logic required to translate the combination of component bypasses that constitute system bypasses.

A manual control switch for each safety system enables the operator to indicate a system's inoperability whenever a component which is not included in the automatic indication system is deliberately bypassed.

The BIS and its logic can be tested by depressing test pushbuttons.

The following systems provide inputs to the Bypass Indication System:

Emergency Service Water System
Diesel Generator Control System
Diesel Generator Output System
Diesel Generator Auxiliary System
Control Room Habitability System
Standby Gas Treatment System
Battery Room Exhaust System
RHR Service Water System
Remote Shutdown Panel
Containment Instrument Gas System
Containment Isolation System
Drywell Ventilation System
Reactor Building Emergency Switchgear And Motor Control Center Cooling
Control Structure HVAC Alternate Operation System

Table 7.5-8 identifies the system and components of the automatic Bypass Indication System.

7.5.1b.8 Post-Accident Neutron Flux Monitoring System

The post-accident neutron flux monitoring function is performed by the Average Power Range Monitoring System (APRM). This system is discussed in Section 7.6.1a.5.6.

The use of the conventional Nuclear Monitoring System (NMS), in particular the APRMs, to perform the post-accident monitoring function was evaluated from an Emergency Procedure Guideline (EPG) standpoint in NEDO-31558A. The results from the BWROG analysis show a

separate accident monitoring system (e.g., the ex-core system) is not required for accident conditions, since the conventional NMS would provide adequate indication. The analysis also showed that, since operators could use other plant parameters to determine neutron flux, even a failure of the conventional NMS would not compromise plant safety.

The NRC Office of Nuclear Reactor Regulation found the BWROG report (NEDO-31558) acceptable and concluded that "Category 1 neutron flux monitoring instrumentation is not needed for existing BWRs to cope with LOCA, ATWAS, or other accidents that do not result in severe core damage conditions. Instrumentation to monitor the progression of core melt accidents would be best addressed by the current severe accident management program."

The ex-core neutron flux monitoring system was originally installed at SSES to meet the requirements of Regulatory Guide 1.97, Revision 2. Based on the results of the BWROG report, the NRC position on this issue, and SSES specific reviews, the post-accident neutron flux monitoring function was changed to be performed by the Average Power Range Monitoring System (APRM). The excore neutron flux monitoring system has been removed from service.

7.5.1b.8.1 (This Subsection Has Been Deleted)

7.5.2a Analysis of NSSS Safety-Related Displays

7.5.2a.1 General

The safety-related display instrumentation provides adequate information to allow the reactor operator to perform the necessary manual safety function.

All protective actions required under accident conditions for the NSSS equipment are automatic, redundant, and decisive such that immediate reactor operator information or intervention is unnecessary.

The ACR design improves the availability of the plant by providing the operator with more readily accessible information and control of the various plant operational parameters. This is accomplished by the logical organization of functional plant system indicators, displays, controls and a computer display system.

A complete description and analysis of design criteria applicable to the hardwired indicators, displays and controls for the various safety-related systems are described elsewhere in Chapter 7 with the systems they serve. Redundancy and independence or diversity are provided in all of those information systems which are used as a basis for operator-controlled safeguards action. A complete failure of the Plant Integrated Computer System which serves as an active part of the operator/plant interface does not degrade the quantity or quality of necessary information presented by hardwired devices needed to determine the status or action of plant safety systems.

7.5.2a.1.1 DESIGN CRITERIA

7.5.2a.1.1.1 Power Generation Control Complex Criteria

The applicable design criteria for the PGCC aspects of the ACR design are provided in General Electric Licensing Topical Report, NEDO-10466-A.

7.5.2a.1.1.2 Advance Control Room Design and Operational Criteria

7.5.2a.1.1.2.1 Design Criteria

(1) The implementation of the ACR design does not affect the ability of any system to meet the requirements of its design specification.

(2) In the implementation of the ACR design, instruments for the reactor protection system and the engineered safety features meet the system design requirements of the systems they serve. They are located at easily visible and accessible positions.

(3) The design employs modular techniques to implement distinct circuits so that separation and redundancy requirements are satisfied.

The interfacing circuitry between the Class 1E safety systems and the non-Class 1E non-safety Display Control System utilizes both digital and analog safety signals. Isolation devices have not been provided between certain Class 1E systems and the non-safety Display Control System.

As an alternative to providing isolation devices, analyses of the Class 1E safety systems from which the non-Class 1E non-safety Display Control System derive their signals was performed. The analyses evaluated the effects of an open or short circuit in the non-safety system component and a fault voltage of 250 VDC (288 VDC max.) on the cabling connecting the non-safety system component to the Class 1E system. The analyses determined that the design of the interfacing circuitry between the Class 1E safety systems and the non-Class 1E non-safety Display Control System was acceptable by demonstrating that the Class 1E safety systems are not degraded below an acceptable level when postulating faults in the non-safety Display Control System that derive their signals from Class 1E circuits. This acceptance is documented in a letter dated November 5, 1997 from the USNRC entitled, “Single Failure Analysis of Class 1E/Non Class 1E Interface Circuits in the GE Scope of Supply, Susquehanna Steam Electric Station, Units 1 and 2 (TAC NOS. M90541 and M90542).”

(4) All reactor protection system components incorporated by the ACR design are of at least comparable quality to the components that are integral to the design of related systems and have demonstrated operational reliability.

(5) The implementation of the ACR makes use of modular control and indication components. Plug-connected cables are used to facilitate removal of the modules. Cables and connectors are easily accessible and identified. Connector separation requires deliberate action.

(6) Cabling is identified at each connection point, in the panels, wireways and termination cabinets so that visual verification of separation can be easily made. Connectors and cabling at connection points are clearly marked with system and reference designations.

(7) All plant system controls remain hardwired. They are external to, and not dependent upon, the computer systems.

(8) Simplification of controls is restricted to manual functions operating independently from, but compatible with, the automatic protective functions.

(9) All safety system functions, either automatic protective or interlocking, including controls, displays and alarms remain hardwired. These system functions can be changed only by physically modifying the wiring or equipment. They are independent from the computer systems.

7.5.2a.1.1.2.2 Operating Criteria

The implementation of the ACR design provides for planned operations or normal plant operation under planned conditions in the absence of significant abnormalities. Operations subsequent to an incident (transient, accident or special event) are not considered planned operations until the procedures being followed or equipment being used are identical to those used during any one of the defined planned operations. The established planned operations can be considered as a chronological sequence from refueling outage to refueling outage. The following planned operations are identified:

a. Refueling Outage
b. Achieving Criticality
c. Heatup
d. Reactor Power Operation
e. Achieving Shutdown
f. Cooldown

7.5.2a.2 Normal Operation

Subsection 7.5.1a.2 describes the basis for selecting ranges for instrumentation. Since abnormal, transient, or accident condition monitoring requirements exceed those for normal operation, the normal ranges are covered adequately. The accuracy of safety-related display instrumentation is included in the Technical Specifications.

7.5.2a.3 Abnormal Transient Occurrences

These occurrences are not limiting from the point of view of instrument ranges and functional capability. (See Subsection 7.5.2a.4.)

The variety of indications which may be utilized to verify that shutdown and isolation safety actions have been accomplished as required (see Subsection 7.5.1a.4.2.3) meets the requirements of IEEE 279-1971.

7.5.2a.4 Accident Conditions

The DBA-LOCA is the most extreme operational event. Information readouts are designed to accommodate this event from the standpoint of operator actions, information, and event tracking requirements, and therefore, will cover all other design basis events or incident requirements. Refer to Chapter 15 "Accident Analysis."

7.5.2a.4.1 Initial Accident Event

The design basis of all engineered safety features to mitigate accident event condition takes into consideration that "no operator action or assistance is required or recommended for the first

twenty (20) minutes of the event with one exception." 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. This requirement therefore makes it mandatory that all protective action necessary in the first twenty minutes be "automatic", excluding the exception above. Therefore, although continuous tracking of variables is available, no operator action based on them is intended.

7.5.2a.4.2 Post-Accident Tracking

The operator has many options which are procedures controlled based on the following information:

(1) Reactor Water Level and Pressure

Vessel water level and pressure instrumentation is redundant and electrically independent. Power is from independent divisionalized Class 1E buses. This instrumentation complies with the independence and redundancy requirements of IEEE 279-1971 and provides recorded outputs. All equipment except the recorders and indicators will perform their required functions during a seismic event.

The reactor water level and pressure sensors are mounted on independent local panels. The sensors and recorders are designed to operate during normal operation and post-accident environmental conditions. The design criteria that the instruments must meet are discussed in Subsection 7.1.2a.1.31.

Refer to Table 7.5-1 during the following discussion.

A combination of three different level instrument ranges, with two independent channels each, monitor Reactor Vessel Level:

1. The extended range instrumentation measure from over the top of the steam lines to near the top of the active fuel and outputs to three control room level indicators per channel.

2. The wide range instrumentation measures from the top of the feedwater control range (just above the high level turbine trip point) down to a point near the top of the active fuel and outputs to one control room recorder and three control room indicators per channel.

3. The fuel zone range instrumentation measures from over the top of the active fuel to near the bottom of the active fuel and outputs to a control recorder (one channel) and a control room indicator (one channel).

These three ranges of instruments combine to provide level indication from the bottom of the Core to above the main steam line.

Two independent channels of reactor vessel pressure (ranged 0 to 1500 psig) are transmitted to and recorded on two separate recorders.

The recorders are located in the control room on the Reactor Core Cooling System benchboard. In the case of the wide range level, one recorder is with the Division 1

systems and the other recorder is with the Division 2 systems. The design, considering the accuracy, range and quality of the instrumentation, is adequate to provide the operator with accurate reactor water level and reactor pressure information during normal operation, abnormal, transient, and accident conditions.

(2) ECCS

Performance of ECCS following an accident may be verified by observing redundant and independent indications as described in Subsection 7.5.1a.4.2.3(3) and fully satisfies the need for operator verification of operation of the system.

(3) Continued Shutdown Tracking

The various indications described in Subsection 7.5.1a.4.2 provide adequate information regarding status of the reactor vessel level and pressure to allow reactor operators to make proper decisions regarding core and containment cooling operations, and fully satisfies the need for post-accident surveillance of these variables.

7.5.2a.4.3 Safe Shutdown Display

The safe shutdown display instrumentation in Subsection 7.5.1a includes the control rod status lamps, scram pilot valve status lamps, and neutron monitoring instrumentation. These displays are expected to remain operable for a long enough time following an accident to indicate the occurrence of safe and orderly shutdown.

The displays provide redundancy by being in three separate systems and the rod position and neutron monitoring outputs are recorded (the former by the process computer). The systems cited are either manually or automatically connectable to the standby AC power.

7.5.2a.4.4 Engineered Safety Feature Operation Display

The other operating instruments covered in Subsection 7.5.1a provide indication of operation of various safety systems but, except for the isolation valve status, do not constitute post-accident surveillance or safe shutdown display. Isolation valve status indication is designed to perform as stated in Subsection 7.5.2a.4.

7.5.2a.5 Compliance with Regulatory/Industry Standards

7.5.2a.5.1 Conformance to IEEE-279-1971

7.5.2a.5.1.1 General

The rod status and scram pilot valve status circuits do not individually meet the requirements of IEEE-279. Jointly, however, they do meet the requirements applicable to display instrumentation except for seismic qualification.

Safety-related portions of the neutron monitoring system are designed to meet all the requirements of IEEE 279 as a part of the RPS. However, its RPS function is a "fail-safe" function while safe shutdown display is not. Further, its RPS function terminates with the generation and maintenance of a shutdown signal. In this regard, post DBA environment conditions may cause malfunction but not until the RPS function of scram generation is concluded. This makes it impossible to claim

continuous indicating capability for safe shutdown display by the neutron monitoring system. Redundancy, power switching capabilities, RPS capabilities, and expected time to failure under DBA environment conditions allow the neutron monitoring system to meet the functional requirements of IEEE-279 as applicable to display instrumentation.

7.5.2a.5.1.2 Automatic Initiation of Protective Action (IEEE-279, Paragraph 4.1)

This requirement is not applicable to the safe shutdown display instrumentation.

7.5.2a.5.1.3 Single Failure Criterion (IEEE 279, Paragraph 4.2)

Indications associated with redundant channels meet the single failure criterion.

7.5.2a.5.1.4 Quality of Indicators (IEEE 279, Paragraph 4.3)

The quality of the indicators will be in accordance with their importance to safety.

7.5.2a.5.1.5 Equipment Qualification (IEEE 279, Paragraph 4.4)

All safety-related equipment except indicators and recorders will be qualified to assure performance of their safety-related functions including post-seismic performance.

7.5.2a.5.1.6 Channel Integrity (IEEE 279, Paragraph 4.5)

The failure of any indicator will not adversely affect channel integrity.

7.5.2a.5.1.7 Channel Independence (IEEE 279, Paragraph 4.6)

The failure of any indicator will not adversely affect channel independence.

7.5.2a.5.1.8 Control and Protection System Interaction (IEEE 279, Paragraph 4.7)

This design requirement is not applicable to the safe shutdown display instrumentation.

7.5.2a.5.1.9 Deviation of System Inputs (IEEE 279, Paragraph 4.8)

This is not applicable to display instrumentation.

7.5.2a.5.1.10 Capability for Sensor Checks (IEEE 279, Paragraph 4.9)

This is not applicable to safe shutdown display instrumentation.

7.5.2a.5.1.11 Capability for Test and Calibration (IEEE 279, Paragraph 4.10)

This is not applicable.

7.5.2a.5.1.12 Channel Bypass (IEEE 279, Paragraph 4.11)

This is not applicable.

7.5.2a.5.1.13 Operating Bypasses (IEEE 279, Paragraph 4.12)

This is not applicable.

7.5.2a.5.1.14 Indication of Bypass (IEEE 279, Paragraph 4.13)

This is not applicable.

7.5.2a.5.1.15 Access to Means for Bypassing (IEEE 279, Paragraph 4.14)

Bypassing is not applicable.

7.5.2a.5.1.16 Multiple Setpoints (IEEE 279, Paragraph 4.15)

This design requirement is not applicable to shutdown instrumentation.

7.5.2a.5.1.17 Completion of Protective Action Once it is Initiated (IEEE 279, Paragraph 4.16)

This is not applicable.

7.5.2a.5.1.18 Manual Actuation (IEEE 279, Paragraph 4.17)

Manual actuation is not applicable to display instrumentation.

7.5.2a.5.1.19 Access to Setpoints (IEEE 279, Paragraph 4.18)

This design requirement is not applicable to display instrumentation.

7.5.2a.5.1.20 Identification of Protective Action (IEEE 279, Paragraph 4.19)

Indicators will indicate protective actions to the channel level.

7.5.2a.5.1.21 Information Read Out (IEEE 279, Paragraph 4.20)

Indicators will provide required information.

7.5.2a.5.1.22 System Repair (IEEE 279, Paragraph 4.21)

This design requirement is not applicable, except for the role played by the indicators in providing diagnostic information.

7.5.2a.5.1.23 Identification (IEEE 279, Paragraph 4.22)

Indicators will be identified.

7.5.2a.5.2 Conformance with IEEE 323

See Subsection 3.11.2.1.

7.5.2a.5.3 Conformance with IEEE 344

See Section 3.10a.2.1.

7.5.2a.5.4 Regulatory Guide 1.47

The Safety-Related Display Instrumentation (SRDI) is designed to operate continuously, and there is no requirement for bypass provisions. Removal of instrumentations for servicing during plant operation is administratively controlled.

7.5.2a.5.5 Regulatory Guide 1.97

Safety-related display instrumentation will be in accordance with PP&L's commitment to Regulatory Guide 1.97, Rev. 2. The systems are discussed in the referenced sections:

- Post-Accident ARMS, see Subsection 12.3.4
- Suppression Pool/Drywell Spray Flow, see Subsection 5.4.7.1

  Standby Liquid Control System, see Subsection 9.3.5.2 Instrument Gas Bottles Pressure Indication, see Subsection 7.3.1.1b.8.3.
- RPV Water Level/Pressure, see Subsection 7.5.1a.4.2
- RHR Heat Exchanger Outlet Temperature, see Subsection 7.4.2.3.1

All other items required for commitments to Regulatory Guide 1.97, Revision 2, appear in the appropriate subsections and tables in Section 7.5.

7.5.2a.5.6 10CFR50, Appendix B

Safety-related display instrumentation environmental qualification is addressed in PP&L's commitment to Regulatory Guide 1.97, Rev. 2.

7.5.2b Analysis of non-NSSS Safety-Related Display Instrumentation

7.5.2b.1 Identification, Redundancy, Accuracy

SRDI which is provided under non-NSSS responsibility is summarized in Subsection 7.5.1b and in Tables 7.5-2 through 7.5-7a. The safety-related displays are for ESF systems and for auxiliary support systems. These tables illustrate the redundancy applied to the displays and location of the panel on which the instrument is mounted, within the plant-operator interface.

7.5.2b.2 Isolation, Separation

Table 1.7-1 lists elementary diagrams which show electrical isolation of divisions. It should be noted that these divisions are routed through separate divisionalized panels and use the divisional separation and identification provisions of the PGCC. Outside PGCC, refer to Section 3.12.

7.5.2b.3 Qualification of Components

Primary containment isolation, primary containment vacuum relief, containment atmosphere, ESW, RHRSW and containment instrument gas rely on NSSS qualifications in Section 3.10a.

Standby gas treatment, reactor building recirculation, habitability systems use qualification described in Section 3.10b. Post-accident monitoring instruments in panel C693, C690A&B, and Bypass Indication are also qualified as described in Section 3.10b.

7.5.2b.4 Capability for Checking

Under normal conditions periodic cross checking between indicators of two divisions, and checking against previous records will provide notification of malfunction. Under any conditions, normal or abnormal, if there is disagreement of readings, it is necessary to check to determine which indication is correct. The suspect indication is checked against a third indictor or checked against an alternate indicator from which the correct indication may be inferred.

7.5.2b.5 Analysis of the Bypass Indication System

The Bypass Indication System (BIS) indicates on panel (C601/H12-P601) that any non-NSSS ESF or ESF supporting system is inoperable. That is indication of inoperability at a system level. Indication of component inoperability within the non-NSSS ESF systems is provided on Panel C694. Both panels are located in the operator interface ring of panels.

Table 7.5-8 lists the systems and components included in the system.

Manual capability for testing operability of each indication is provided. The system design maintains the divisionalized structure of the ESF and signals to the BIS are mechanically and electrically isolated from the associated ESF system.

Regulatory Guide 1.47 and Branch Technical Position E1CSB 21 are complied with in the design of BIS.

7.5.2b.6 This Section Has Been Deleted

TABLE 7.5-1
SAFETY RELATED DISPLAY INSTRUMENTATION

Design Criteria	Type Readout[#]		Panel[#] Number	Number of Channels	Rated Range	Location	PAM
Reactor Vessel Pressure	Recorder	UR14201A/B	1C601	2	0-1500 PSIG	CR	X
	Indicator	PI14202A/B	1C601	2	0-1500 PSIG	CR	X
		PI14204A/B	1C601	2	0-1500 PSIG	CR	X
		PI14202A1/B1	1C651	2	0-1500 PSIG	CR	X
		PI14262	1C201	1	0-1500 PSIG	RSP	X
Reactor Vessel Water Level	Recorder	UR14201A/B	1C601	2	-150"/0/+60" (wide)	CR	X
	Recorder	UR14201a	1C601	1	-310"/TAF/-110" (fuel zone)	CR	X
	Indicator	LI14201A/B	1C601	2	-150"/0/+180" (extended) -150"/0/+60" (wide)	CR	X
	Indicator	LI14203A	1C601	2	-150"/0/+180" (extended) -150"/0/+60" (wide)	CR	X
	Indicator	LI14201A1/B1	1C651	2	-150"/0/+180" (extended) -150"/0/+60" (wide)	CR	X
	Indicator	UR14201B	1C601	1	-310"/TAF/-110" (fuel zone) -150"/0/+60" (wide)	CR	X
		LI14262	1C201	1		RSP	
RCIC Flow	Indicator	FIE511R600-1	1C601	1	0-700 GPM	CR	X
RCIC Discharge Pressure	Indicator	PIE511R601	1C601	1	0-1500 PSIG	CR	
HPCI Flow	Indicator	FIE411R600-1	1C601	1	0-6000 GPM	CR	X
HPCI Discharge Pressure	Indicator	PIE411R601	1C601	1	0-1800 PSIG	CR	
HPCI Turbine Steam Pressure	Indicator	PIE411R602	1C601		0-1500 PSIG	CR	
CS Flow	Indicator	FIE211R601A	1C601	2	0-10,000 GPM	CR	X
RHR Flow (LPCI and Shutdown Cooling)	Indicator	FIE111R603A	1C601	2	0-30,000 GPM	CR	
RHR Service Water Flow	Indicator	FIE111R602A	1C601	2	0-12,000 GPM	CR	
RHR Heat Exchanger Outlet Temperature	Indicator	TI15127A	1C601	2	40-340°F	CR	X
Primary Containment Area Radiation	Recorder	RR15755A	1C693	2	10-10E6 CPM	CR	X

#: The indicator/recorder numbers and panel numbers shown are for Unit 1. The Unit 2 numbers are the same except that a 2 appears before the sequence number instead of a 1.

TABLE 7.5-2

SAFETY RELATED DISPLAY INSTRUMENTATION
HABITABILITY (EMERGENCY OUTSIDE AIR SYSTEM/CONTROL STRUCTURE HVAC)

Parameter Measured	No. of Channels	Range	Accuracy	Type of Readout	Location	Panel No.	Power IE Bus	RPS	ESF	AS	PPD	Remarks
CS NORM OA SUPPLY DMP HDD7802A	2			LT	CR	0C681	Yes		X			
CREOASS A,B INLET DMP HDD7814A	2			LT	CR	0C681	Yes		X			
CREOASS A,B DSCH DMP HDD7811A	2			LT	CR	0C681	Yes		X			
EOASS RET AIR DMP HDD7813A	2			LT	CR	0C681	Yes		X			
CR RELIEF ISO DMP HDD7833A	2			LT	CR	0C681	Yes		X			
CREOASS AIRFLOW A/B FIC07816A	2	0-10,000 cfm	± 2%	Ind/Ctr	CR	0C681	Yes		X			
BATTERY RMS EXH SYS DMP HDD787182	4			LT	CR	0C681	Yes		X			

Note: PAM = Post Accident Monitoring; RSP = Remote Shutdown Panel; RPS = Reactor Protection System; ESF = Engineered Safety Feature; AS = Auxiliary Support; PPD = Plant Process Display

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TABLE 7.5-3

SAFETY-RELATED DISPLAY INSTRUMENTATION
CONTAINMENT AND SUPPRESSION POOL INSTRUMENTATION

Parameter Measured	No. of Channels	Range	Accuracy	Type of Readout	Location	Panel No.	Power IE Bus	RPS	ESF	AS	PAM	PPD	Remarks
Containment Hydrogen	2	0-10% 0-30%	±2%	Ind	CR	C690	Yes		X		X	X	
Containment Oxygen	2	0-10% 0-25%	±2%	Ind	CR	C690	Yes		X		X	X	0-25% Range is not calibrated
Containment O_2/H_2	2	0_2:0-10% 0-25% H_2:0-10% 0-30%	±2%	Rec	CR	C601	Yes		X		X	X	Operating History 0-25% Range is not calibrated
Drywell Temperature	2	50-350°F	±2%	Ind-Rec	CR	C693	No				X	X	Operating History
Containment Temperature	2	40-440°F	±2%	Rec	CR	C601	Yes				X		
Containment Temperature	1	0-350°F	±2%	Ind	RSP	C201	Yes					X	
Suppression Pool Water Temperature	2	30-230°F	±2%	Ind-Rec	CR	C690	Yes			X	X	X	
Suppression Pool Water Temperature	2	30-230°F	±2%	Ind	RSP	C201	Yes					X	
Suppression Pool Water Temperature	2	30-230°F	±2%	Ind	CR	C601	Yes						
Suppression Chamber Atmos Temperature	2	0-400°F	±5%	Ind-Rec	CR	C693	No					X	
Suppression Pool Air Temperature	1	0-350°F	±2%	Ind	RSP	C201	Yes					X	
Drywell Hi Accident Range Pressure	2	0-250 psig	±2%	Rec	CR	C601	Yes				X	X	Operating History

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Parameter Measured	No. of Channels	Range	Accuracy	Type of Readout	Location	Panel No.	Power IE Bus	RPS	ESF	AS	PAM	PPD	Remarks
Drywell LOCA Range Pressure	2	-14.7 to 65 psig	±2%	Rec	CR	C601	Yes				X	X	Operating History
Drywell Normal Range Pressure	1	-3 to +3 psig	±2%	Ind	CR	C601	No					X	Normal Operation
Drywell Normal Range Pressure	1	-3 to +3 psig	±2%	Ind	RSP	C201	Yes					X	Normal Operation
Suppression Chamber LOCA Range Pressure	2	-14.7 to +65 psig	±2%	Rec	CR	C601	Yes					X	Operating History
Suppression Chamber Normal Range Pressure	1	-3 to +3 psig	±2%	Ind	CR	C601	No					X	Normal Operation
Suppression Pool Level (wide range)	2	5-49 feet	±5%	Rec	CR	C601	Yes			X	X		
Suppression Pool Level (narrow range)	2	18-26.5 feet	±5%	Rec	CR	C601	Yes				X	X	Operating History
Suppression Pool Level (narrow range)	2	18-26.5 feet	±2%	Ind	CR	C601	Yes				X	X	
Suppression Pool Level (wide range)	-*	4.5-49 feet	±2%	Ind	RSP	C201	Yes			X			Remote Shutdown
Containment & Suppression Chamber Gas Sample Vlv's Inboard/Outboard Isolation Status	2	---		LT	CR	C601	Yes		X		X	X	
Containment & Suppression Chamber Nitrogen Purge Inboard/Outboard Isolation Vlv Status	2	---		LT	CR	C601	Yes		X		X	X	
Suppression Chamber Spray Flow	2	0-750 GPM	±1%	Ind	CR	C601	Yes						
Drywell Spray Flow	2	0-12000	±1%	Ind	CR	C601	Yes						

NOTE: PAM = Post Accident Monitoring RSP = Remote Shutdown Panel RPS = Reactor Protection System
ESF = Engineered Safety Feature AS = Auxiliary Support PPD = Plant Process Display

TABLE 7.5-4

SAFETY RELATED DISPLAY INSTRUMENTATION
STANDBY GAS TREATMENT SYSTEM, RX BLDG RECIRCULATION AND ISOLATION SYSTEM

Parameter Measured	No. of Channels	Range	Accuracy	Type of Readout	Location	Panel No.	Power IE Bus	RPS	ESF	AS	PPD	Remarks
RB Zone 1 Outdoor (Pressure Diff.)	2	0.2-1 in wg	±2%	Ind	CR	0C681	Yes		X			
RB Zone 2 Outdoor (Pressure Diff.)	2	0.2-1 in wg	±2%	Ind	CR	0C681	Yes		X			
RB Zone 3 Outdoor (Pressure Diff.)	2	0.2-1 in wg	±2%	Ind	CR	0C681	Yes		X			
RB Zones Outdoor (Lowest Pressure Diff.)	2	0.2-1 in wg	±2%	Ind/Ctlr	CR	0C681	Yes		X			
SGTS Unit 1 Drywell & Wetwell Damper A	1			LT	CR	1C681	Yes		X			
SGTS Unit 1 Drywell & Wetwell Damper B	1			LT	CR	1C681	Yes		X			
SGTS Unit 2 Drywell & Wetwell Damper A	1			LT	CR	2C681	Yes		X			
SGTS Unit 2 Drywell & Wetwell Damper B	1			LT	CR	2C681	Yes		X			
SGTS Flow	2	0-15,000 cfm	±2%	Ind/Ctlr	CR	0C681	Yes		X			
RB Recirc System to Zone 1 Supply A,B	2			LT	CR	1C681	Yes		X			
RB Recirc System to Zone 1 Exhaust A,B	2			LT	CR	1C681	Yes		X			
RB Recirc System to Zone 1 Equip. Compartment Exh. A,B	2			LT	CR	1C681	Yes		X			

TABLE 7.5-4

SAFETY RELATED DISPLAY INSTRUMENTATION
STANDBY GAS TREATMENT SYSTEM, RX BLDG RECIRCULATION AND ISOLATION SYSTEM

Parameter Measured	No. of Channels	Range	Accuracy	Type of Readout	Location	Panel No.	Power IE Bus	RPS	ESF	AS	PPD	Remarks
Recirc Pump A,B Motor Cooler Inlet Outboard Isol. Valve	2			LT		1C681	Yes		X			
SGTS Outdoor/Zone 1 Pressure Diff. A,B	2	0.2-1 in wg	±2%	Ind	CR	0C681	Yes		X			
SGTS Outdoor/Zone 2 Pressure Diff. A,B	2	0.2-1 in wg	±2%	Ind	CR	0C681	Yes		X			
SGTS Outdoor/Zone 3 Pressure Diff. A,B	2	0.2-1 in wg	±2%	Ind	CR	0C681	Yes		X			
SGTS Outdoor/Zones Pressure Diff. A,B	2	0.2-1 in wg	±2%	Ind/Ctlr	CR	0C681	Yes		X			
SGTS Unit 1 Drywell & Wetwell Damper A	1			LT	CR	1C681	Yes		X			
SGTS Unit 1 Drywell & Wetwell Damper B	1			LT	CR	1C681	Yes		X			
SGTS Unit 2 Drywell & Wetwell Damper A	1			LT	CR	2C681	Yes		X			
SGTS Unit 2 Drywell & Wetwell Damper B	1			LT	CR	2C681	Yes		X			
SGTS Flow	2	0-15,000 cfm	±2%	Ind/Ctlr	CR	0C681	Yes		X			
RB Recirc System to Zone 1 Supply A,B	2			LT	CR	1C681	Yes		X			
RB Recirc System to Zone 1 Exhaust A,B	2			LT	CR	1C681	Yes		X			

TABLE 7.5-4

SAFETY RELATED DISPLAY INSTRUMENTATION
STANDBY GAS TREATMENT SYSTEM, RX BLDG RECIRCULATION AND ISOLATION SYSTEM

Parameter Measured	No. of Channels	Range	Accuracy	Type of Readout	Location	Panel No.	Power IE Bus	RPS	ESF	AS	PPD	Remarks
RB Recirc System to Zone 1 Equip. Compartment Exh. A,B	2			LT	CR	1C681	Yes		X			
Recirc Pump A,B Motor Cooler Inlet Outboard Isol. Valve	2			LT		1C681	Yes		X			
Recirc Pump A,B Motor Cooler Outlet Outboard Isol. Valve	2			LT		1C681	Yes		X			
Recirc Loop A,B Incoming Line Outboard Isol. Valve	2			LT		1C681	Yes		X			
Recirc Loop A,B Outgoing Line Outboard Isol. Valve	2			LT		1C681	Yes		X			

Note: PAM = Post Accident Monitoring; RSP = Remote Shutdown Panel; RPS = Reactor Protection System; ESF = Engineered Safety Feature; AS = Auxiliary Support; PPD = Plant process Display

TABLE 7.5-5

SAFETY RELATED DISPLAY INSTRUMENTATION
EMERGENCY SERVICE WATER SYSTEM

Parameter Measured	No. of Channels	Range	Accuracy	Type of Readout	Location	Panel No.	Power IE Bus	RPS	ESF	AS	PAM	PPD	Remarks
ESW Flow Loop A/B	2	0-12,000 GPM	±2%	Ind/Rec	CR	C653	Yes			X	X		
ESW Pump AC/BD Discharge Pressure	2	0-200 psig	±2%	Ind	CR	C653	Yes			X			
ESW Pump AC/BD Discharge Temperature	2	0-200° F	±2%	Ind	CR	C653	Yes			X	X		
ESW Diesel Cooler Outlet Temperature	4	50-150° F	±2%	Ind	CR	C653	No			X			
ESW and RHRSW Total Flow A/B	2	0-30,000 GPM	±0.5%	Rec	CR	C653	No			X			
ESW Pump ABCD Status	4	---	---	LT	CR	C653	Yes			X			
ESW Diesel Cooler Inlet/Outlet Valve Status	8	---	---	LT	CR	C653	Yes			X			
Diesel Generator Room Flooded	4	---	---	LT	CR	C653	Yes			X			
ESW Structure A/B Flooded	2	---	---	Annun	CR	C653	Yes			X			
ESW Loop A/B in Service	2	---	---	LT	CR	C653	Yes			X			

Note: PAM = Post Accident Monitoring; RSP = Remote Shutdown Panel; RPS = Reactor Protection System; ESF = Engineered Safety Feature; AS = Auxiliary Support; PPD = Plant Process Display

TABLE 7.5-6

SAFETY RELATED DISPLAY INSTRUMENTATION
RHR SERVICE WATER SYSTEM AND SPRAY POND

Parameter Measured	No. of Channels	Range	Accuracy	Type of Readout	Location	Panel#	Power IE Bus	RPS	ESF	AS	PAM	PPD	Remarks
RHRSW HX A Inlet Flow	1	0-12,000 GPM	±2%	Ind	CR	C601	yes			X			
RHRSW HX B Inlet Flow	1	0-12,000 GPM	±2%	Ind	CR	C601	yes			X			
RHRSW HX A Inlet Flow (Unit 2 only)	1	0-12,000 GPM	±2%	Ind	ESP	C201	yes			X			Remote Shutdown
RHRSW HX B Inlet Flow (Unit 1 only)	1	0-12,000 GPM	±2%	Ind	ESP	C201	yes			X			Remote Shutdown
RHRSW Pump A,B Status	2	---	---	LT	CR	C601	yes			X			
RHRSW HX A/B Inlet Temperature	2	0-100°F	±2%	Ind	CR	C601	no					X	
RHRSW HX A/B Inlet Vlv Position	2	0-100%	±2%	Ind	CR	C601	no			X			Status on PSP
RHRSW HX A/B Dish Vlv Status	2	---	---	LT	CR	C601	yes			X			Status on PSP
RHRSW Loop A/B Flow to Spray Pond	2	0-20,000 GPM	±.5%	Rec	CR	C653	no						
RHRSW Pump A/B Disch Pressure	2	0-200 PSIG	±2%	Ind	CR	C601	no					X	
RHRSW Crosstie A/B to RHR Vlv Status	4	---	---	LT	CR	C601	yes			X		X	
ESS Spray Pond Inlet Vlv Loop A/B Status	2	---	---	LT	CR	C653	yes			X			
ESS Spray Pond Bypass Vlv Loop A/B Status	2	---	---	LT	CR	C653	yes			X			
ESS Spray Pond Temperature	4	25-125°F	±2%	Ind	CR	C653	no					X	Selective Channels

Notes:PAM = Post Accident Monitoring; RSP = Remote Shutdown Panel; RPS = Engineered Safety Feature; AS = Auxiliary Support; PPD = Plant Process Display

TABLE 7.5-7

SAFETY RELATED DISPLAY INSTRUMENTATION
CONTAINMENT INSTRUMENT GAS SYSTEM

Parameter Measured	No. of Channels	Range	Accuracy	Type of Readout	Location	Panel No.	Power IE Bus	RPS	ESF	AS	PAM	PPD	Remarks
Instrument Gas Pressure to Main Steam Relief Valve	1	0-200 psig	±2%	Ind	CR	C601	No			X			
Instrument Gas Supply Pressure	1	0-100 psig	±2%	Ind	CR	C601	No			X			
Instrument Gas Bottle Pressure Indication (Unit 2 only)	2	0-3000 psig	±2%	Ind	CR	C601	Yes			X	X		Computer Input
Instrument Gas Bottles Isol Vlv Status	2	---	---	LT	CR	C601	Yes			X			*)
Instrument Gas Suction IB/OB Isol Vlv Status	2	---	---	LT	CR	C601	Yes			X			*)
Instrument Gas Containment IB/OB Isol Vlv Status	9	---	---	LT	CR	C601	Yes			X			*)

Note: PAM = Post Accident Monitoring; RSP = Remote Shutdown Panel; RPS = Reactor Protection System; ESF = Engineered Safety Feature; AS = Auxiliary Support; PPD = Plant Process Display

TABLE 7.5-7a

SAFETY-RELATED DISPLAY INSTRUMENTATION
STANDBY LIQUID CONTROL SYSTEM

Parameter Measured	No. of Channels	Range	Accuracy	Type of Readout	Location	Panel No.	Power IE Bus	RPS	ESF	AS	PAM	PPD	Remarks
SLCS Flow	1	0-100 GPM	±2%	Indicator	14806 CR	C601	Yes				X		
SLCS Storage Tank Level	1	0-5000 Gal	±2%	Indicator	14806 CR	C601	Yes				X		

TABLE 7.5-8

BYPASS INDICATION FOR NON-NSSS SYSTEMS

Bypassed System Indication	Bypassed Component Indication
ESW, Loop A Out-of-Service	ESW Pump 0P-504A Power Loss ESW Pump 0P-504C Power Loss ESW Pump Cooling Fans A/C Disabled
ESW, Loop B Out-of-Service	ESW Pump 0P-504B Power Loss ESW Pump 0P-504D Power Loss ESW Pump Cooling Fans B/D Disabled
Diesel-Gen. A(B,C,D) Control System Out-of-Service	DG A(B,C,D) DC Control Power Loss DG A(B,C,D) Field Flash and Excitation Control Power Loss DG A(B,C,D) Auto Control Unavailable DG A(B,C,D) Cooling Fans Disabled 4 Kv Bus 1AXFMR101CRT Breaker Disabled
Diesel-Gen. A(B,C,D) Output System Out-of-Service	DG A(B,C,D) Breaker Racked Out DG A(B,C,D) Control Power Loss 4 Kv Bus 1A XFMR 201CRT Breaker Disabled
Diesel-Gen. A(B,C,D) Aux. System Out-of-Service	DG A(B,C,D) Aux. Supply/Control Power Loss DG A(B,C,D) Aux. System Not in Auto Oil Pump 0P-514A(B,C,D) Disabled
Control Room Habitability System Div. I Out-of-Service	Fan 0V-101A Disabled Fan 0V-103A Disabled Fan 0V-115A Disabled Fan 0V-117A Disabled Chilled Water Disabled
NOTE: Whenever Diesel Generator 'E' is aligned for Diesel Generator A, B, C, or D, the Diesel Generator 'E' Bypass Indications are transferred into the Bypass Indication System in place of the substituted diesel generator.	
Control Room Habitability System Div. II Out-of-Service	Fan 0V-101B Disabled Fan 0V-103B Disabled Fan 0V-115B Disabled Fan 0V-117B Disabled Chilled Water Disabled
Standby Gas Treatment System Div. I Out-of-Service	Fan 0V-109A Disabled Fan 0V-118A Disabled Fan 0V-144A Disabled Fan 0V-201A Disabled

TABLE 7.5-8

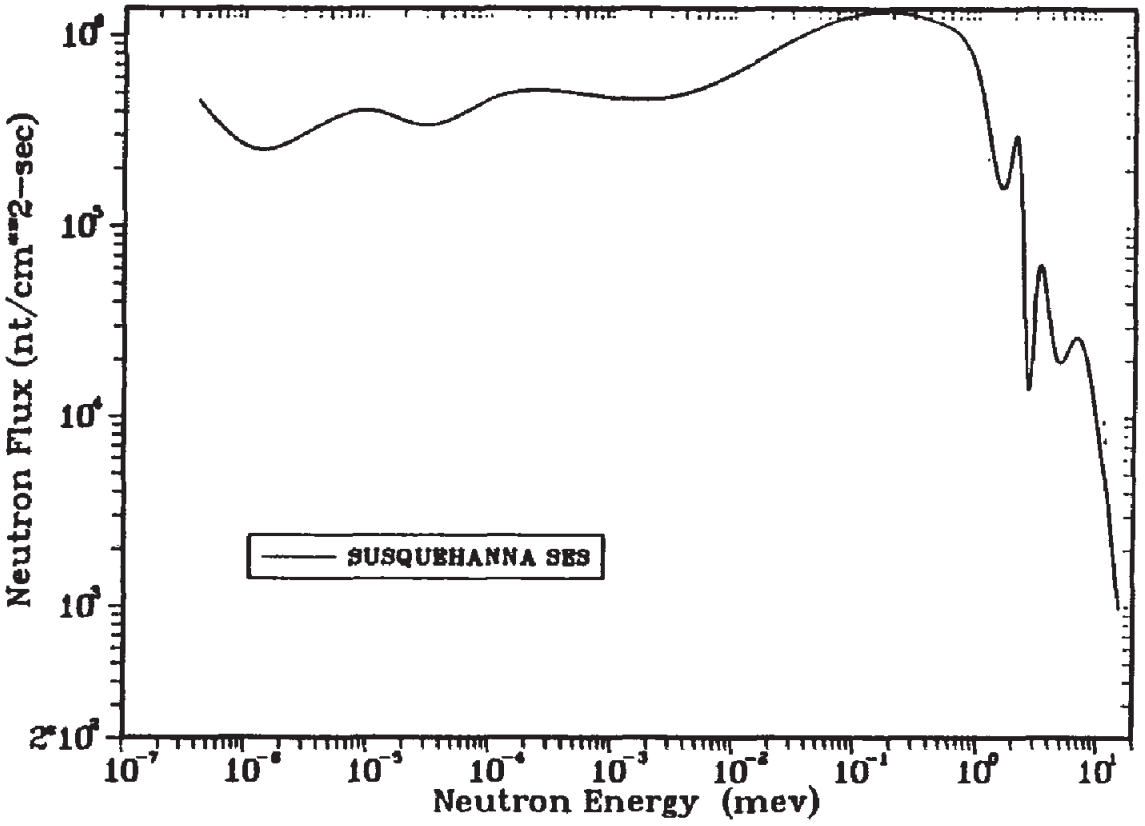
BYPASS INDICATION FOR NON-NSSS SYSTEMS

Bypassed System Indication	Bypassed Component Indication
Standby Gas Treatment System Div. II Out-of-Service	Fan 0V-109B Disabled Fan 0V-118B Disabled Fan 0V-144B Disabled Fan 0V-201B Disabled
Battery Room Exhaust Div. I Out-of-Service	Fan 0V-116A Disabled
Battery Room Exhaust, Div. II Out-of-Service	Fan 0V-116B Disabled
RHR Service Water Loop A Out-of-Service	Pump 1P-506A Power Loss Heat Exchanger Valve Control Power Loss Fan 1V-506A Disabled Control Power Loss for Spray Pond Network Valves Spray Pond Drain Valves Power Loss / Not In Auto
RHR Service Water Loop B Out-of-Service	Pump 1P-506B Power Loss Heat Exchanger Valve Control Power Loss Fan 1V-506B Disabled Control Power Loss for Spray Pond Network Valves Spray Pond Drain Valves Power Loss / Not In Auto
Remote Shutdown Panel Div. I Switches in Emergency Position	Remote Shutdown Panel Div. I Switches in Emergency Position
Remote Shutdown Panel Div. II Switches in Emergency Position	Remote Shutdown Panel Div. II Switches in Emergency Position
Containment Instrument Gas System Div. II Out-of-Service	Nitrogen Supply HV-12648 Open Nitrogen Supply HV-12643 Open
Containment Isolation System Div. I Out-of-Service	Loss of Control Power for Motorized Valves (Div. I) Containment Atmos Purge ISO Bypass
Containment Isolation System Div. II Out-of-Service	Loss of Control Power for Motorized Valves (Div. II) Containment Atmos Purge ISO Bypass

TABLE 7.5-8

BYPASS INDICATION FOR NON-NSSS SYSTEMS

Bypassed System Indication	Bypassed Component Indication
	Unit 1 Unit 2
Drywell Ventilation System Div. I Out-of-Service	Fan 1V-414A / 2V414A Disabled Fan 1V-418A / 2V415A Disabled Fan 1V-416A / 2V416A Disabled
Drywell Ventilation System Div. II Out-of-Service	Fan 1V-414B / 2V414B Disabled Fan 1V-418B / 2V415B Disabled Fan 1V-416B / 2V416B Disabled
Reactor Building Emerg. Switchgear and LCC Cooling Div. I Out-of-Service	Fan 1V-222A Disabled
Reactor Building Emerg. Switchgear and LCC Cooling Div. II Out-of-Service	Fan 1V-222B Disabled
Control Structure HVAC Train 'A' Alternate Control Switches in Emergency Position	CSHVAC Alternate Control Panel Switches in Emergency Position



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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
NEUTRON FLUX AT PRIMARY FIELD FIGURE 7.5-3, Rev 49

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