CHAPTER 7

INSTRUMENTATION AND CONTROL

7.1 GENERAL DESIGN CRITERIA

Complete supervision of both the nuclear and turbine-generator sections of the plant is accomplished by the instrumentation and control systems from the control room. The instrumentation and control systems are designed to permit periodic on-line test to demonstrate the operability of the reactor protection system.

Criteria applying in common to all instrumentation and Control Systems are given in Section 7.1.1. Thereafter, criteria which are specific to one of the instrumentation and control systems are discussed in the appropriate portion of the description of that system, as referenced in Section 7.1.2.

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

[Historical Information] The Authority has completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

7.1.1 Instrumentation and Control Systems Criteria

Instrumentation and Control Systems

Criterion: Instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential reactor facility operating variables. (GDC 12 of 7/11/67)

Instrumentation and controls essential to avoid undue risk to the health and safety of the public are provided to monitor and maintain neutron flux, primary coolant pressure, flow rate, temperature, and control rod positions within prescribed operating ranges.

The non-nuclear regulating process and containment instrumentation measures temperatures, pressure, flow, and levels in the Reactor Coolant System, Steam Systems, Containment and other Auxiliary Systems.

Process variables required on a continuous basis for the startup, power operation, and shutdown of the plant are controlled form and indicated or recorded at the control room, access to which is supervised. The quantity and types of process instrumentation provided ensure safe and orderly operation of all systems and processes over the full operating range of the plant.

7.1.2 Related Criteria

The following are criteria which are related to all instrumentation and control systems but are more specific to other plant features or systems, and therefore are discussed in other chapters, as listed.

Title of Criterion (7/11/67 issue)	Reference
Suppression of Power Oscillations (GDC 7)	Chapter 3
Reactor Core Design (GDC 6)	Chapter 3
Quality Standards (GDC 1)	Chapter 4
Performance Standards (GDC 2)	Chapter 4
Fire Protection (GDC 3)	Chapter 5 and 9
Missile Protection (GDC 40)	Chapters 4, 5, and 6
Emergency Power (GDC 39 and GDC 24)	Chapter 8

7.2 PROTECTIVE SYSTEMS

The protective systems consist of both the Reactor Protection System and the Engineered Safety Features. Equipment supplying signals to any of these protective systems is considered a part of that protective system.

7.2.1 Design Bases

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently, made a part of 10 CFR 50.

The Authority has completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NCR) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980 and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

Control Room

Criterion: The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit continuous occupancy of the control room under any credible post-accident condition or as an alternative, access to other areas of the facility as necessary to shut down and maintain safe control of the facility without excessive radiation exposure of personnel. (GDC 11 of 7/11/67)

Indian Point 3 is equipped with a Control Room which contains those controls and instrumentation necessary for operation of the reactor and turbine generator under normal and accident conditions.

The Control Room is provided with emergency lighting; color coding, labeling and demarcation of reactor coolant control and display panels; switch protection; and other aids as required to ensure proper operation of the reactor, turbine generator and auxiliaries under all operating and accident conditions.

The Control Room is continuously occupied by qualified operating personnel under all operating and Maximum Credible Accident (MCA) conditions. The Post Accident Monitoring instrumentation available to the operator for monitoring plant conditions is provided in Table 7.5-1. The instrumentation complies with Regulatory Guide 1.97 requirements, as documented in NRC Letter, J.D. Neighbors to R. Beedle, dated 4/3/91, entitled "Emergency Response Capability – Conformance To RG 1.97 Revision 3, for Indian Point 3" (TAC No. 51099).

The instrumentation originally available to the operator for monitoring conditions in the Reactor, Reactor Coolant System and the Containment Building are provided in Historical Tables 7.2-4 and 7.2-5.

Historical Table 7.2-4 lists indication (meters, recorders, etc.) available for providing information following moderate and infrequent faults as originally analyzed in Chapter 14. Similarly, Historical Table 7.2-5 relates to limiting faults such as a LOCA as originally analyzed in Chapter 14.

The design criteria used in the selection of the original readouts were:

1) The range of readouts extend over the maximum expected range of the variable being measured as a result of faults originally analyzed in Chapter 14.
2) The combined indicated accuracies are within the errors originally assumed in the safety analysis.

Sufficient shielding, distance, and containment integrity are provided to assure that control room personnel shall not be subjected to doses under postulated accident conditions during occupancy of, ingress to and egress from the Control Room which, in the aggregate, would exceed that limits in 10 CFR 100. The control room ventilation consists of a system having a large percentage of recirculated air. The fresh air intake can be closed automatically or by manual backup to stop the intake of airborne activity if monitors indicate that such action is appropriate.

Core Protection Systems

Criterion: Core protection systems, together with associated equipment, shall be designed to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits. (GDC 14 of 7/11/67)

The basic reactor tripping philosophy is to define a region of power and coolant temperature conditions allowed by the primary tripping functions, the overpower ΔT trip, the overtemperature ΔT trip and the nuclear overpower trip. The allowable operating region within these trip settings is provided to prevent any combination of power, temperatures and pressure which would result in DNB with all reactor coolant pumps in operation. Additional tripping functions such as a high pressurizer pressure trip, low pressurizer pressure trip, high pressurizer water level trip, loss of flow trip, steam and feed-water flow mismatch trip, steam generator low-low water level trip,

turbine trip, safety injection trip, nuclear source and intermediate range trips, and manual trip are provided as backup tot he primary tripping functions for specific accident conditions and mechanical failures.

A dropped rod signal blocks automatic rod withdrawal and also provides a turbine load cutback if above a given power level. The dropped rod is indicated from individual rod position indicators or by a rapid flux decrease on any of the power range nuclear channels.

Over power ΔT, overtemperature ΔT, and T_{avg} deviation rod stops prevent abnormal power conditions which could result from excessive control rod withdrawal initiated by a malfunction of the Reactor Control System or by operator violation of administrative procedures.

Engineered Safety Features Protection Systems

Criterion: Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features (GDC 15 of 7/11/67).

Instrumentation and controls provided for the protective systems are designed to trip the reactor in order to prevent or limit fission product release from the core, and to limit energy release, to signal containment isolation, and to control the operation of engineered safety features equipment.

The Engineered Safety Features are actuated by the engineered safety features actuation channels. Each coincidence network energizes an engineered safety features actuation device, which operates the associated engineered safety features equipment, motor starters and valve operators. The channels are designed to combine redundant sensors, independent channel circuitry, coincident trip logic and different parameter measurements so that a safe and reliable system is provided in which a single failure will not defeat the protective function. The action initiating sensors, bistables and logic is shown in the figures which are included in the detailed engineered safety features instrumentation description given in the design section for each system. The engineered safety features instrumentation system actuates (depending on the severity of the condition) the Safety Injection System, the Containment Isolation System, the Containment Air Recirculation System, and the Containment Spray System.

The passive accumulators of the Safety Injection System do not require signal or power sources to perform their function. The actuation of the active portion of the Safety Injection System is described later in this section.

The containment air recirculation coolers are normally in use during plant operation. These units are, however, in the automatic sequence, which actuates the engineered safety features upon receiving the necessary actuating signals indicating an accident condition. The fan cooler bypass valves open on a safety injection signal to provide maximum service water flow.

Containment spray is actuated by coincident and redundant high containment pressure signals (high-high level).

The Containment Isolation System provides the means of isolating the various pipes passing through the containment walls as required to prevent the release of radioactivity to the outside environment in the event of a Loss-of-Coolant Accident.

Protection Systems Reliability

Criterion: Protection systems shall be designed for high functional reliability and in-service testability necessary to avoid undue risk to the health and safety of the public. (GDC 10 of 7/11/67)

The reactor uses the high speed version of the Westinghouse magnetic-type control rod drive mechanisms. Upon a loss of power to the coils, the Rod Cluster Control (RCC) assemblies with full length absorber rods are released and fall by gravity into the core.

The reactor internals, fuel assemblies and drive system components were designed as seismic Class I equipment. The RCC assemblies are fully guided through the fuel assembly and for the maximum travel of the control rod into the guide tube. Furthermore, the RCC assemblies are never fully withdrawn from their guide thimbles in the fuel assembly. For this reason, and because of the flexibility designed into the RCC assemblies, abnormal loadings and misalignments can be sustained without impairing operation of the RCC assemblies.

The Rod Cluster Control assembly guide system is locked together with pins throughout its length to ensure against misalignments which might impair control rod movement under normal operating conditions and credible accident conditions. An analogous system has successfully undergone 4132 hours of testing in the Westinghouse Reactor Evaluation Center during which about 27,200 feet of step-driven travel and 1461 trips were accomplished with test misalignments in excess of the maximum possible misalignment experienced when installed in the plant.

All primary reactor trip protection channels required during power operation are supplied with sufficient redundancy to provide the capability for channel calibration and test at power.

Removal of one trip circuit is accomplished by placing that circuit in a tripped mode i.e., a two-out-of-three circuit becomes a one-out-of-two circuit. A Channel bistable may also be placed in a bypassed mode, i.e., a two-out-of-three circuit becomes a two-out-of-two circuit. Testing in a bypassed mode does not trip the system even if a trip condition exists in a concurrent channel.

Reliability and independence are obtained by redundancy within each tripping function. In a two-out-of-three circuit, for example, the three channels are equipped with separate primary sensors. Each channel is continuously fed from its own independent electrical source. Failure to de-energize a channel when required would be a mode of malfunction that would affect only that channel. The trip signal furnished by the two remaining channels would be unimpaired in this event.

Protection Systems Redundancy and Independence

Criterion: Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure on removal from service of any component or channel of such a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection function to be served. (GDC 20 of 7/11/67)

The Reactor Protection Systems were designed so that the most probable modes of failure (loss of voltage, relay failure) in each protection channel result in a signal calling for the protective

trip. Each protection system design combines redundant sensors and channel independence with coincident trip philosophy so that a safe and reliable system is provided in which a single failure will not defeat the channel function, cause a spurious plant trip, or violate reactor protection criteria.

The design basis for the Reactor Protection System and Engineered Safety Features equipment radiation exposure was that the equipment must function after the exposure associated with the TID-14844 model accident. The maximum anticipated exposure for components located within the Containment was calculated to be 1.6 x 10^8 rads, which is accumulated during one year following the accident. (Note that the integrated exposure for safeguards equipment during 40 years of operation was calculated to be less than 5 x 10^5 rads.) In the determination of exposure, no credit was taken for containment cleanup or other removal mechanism other than isotope decay. The expected integrated exposure on the outside of the Containment Building, again assuming TID-14844 releases and no credit for cleanup, will be less than 10^2 rads integrated over a year at the containment outside surface.

Protection system instrument cables are divided into four channels. Channeling separation is continuous from instrument sensor to receiver. Bistable or digital type outputs 120 volts AC or 125 volts DC to protection system logic relays are divided into the same four channels.

Power and control cables for engineered safeguards are divided into three basic channel systems. Power and control cabling for reactor trip and containment isolation valves are divided into two channels.

In addition to channels of separation, cables were assigned to individual routing systems in accordance with their voltage level, size, and function. Six independent conduit and tray systems are employed on Indian Point 3 as follows:

1) 6900 volt power
2) Heavy 125 volts DC power cables and heavy 480 volts AC (over 100 hp) power cables
3) Lighting panel feeders and medium power (greater than No 12 AWG wire size) 480 volts AC cables
4) Control and light (non-heavy) power cables
5) Instrument cables
6) Rod control cables

Conduit fill for all systems is based on standard national Electric Code Recommendations. Criteria for tray fill are given in Section 8.2

Cables in the conduit and cable schedule are identified by a circuit code, in addition to their routing, to assure that the cable will be installed in the proper tray systems, as well as the proper channel.

Separation of channels was established throughout the plant by the use of separate trays or conduits (exceptions are documented and justified in Reference 1). In addition, whenever a heavy power tray was located less than three feet beneath any tray of a different channel, a transit fire barrier was installed between the trays. A vertical barrier was installed where trays of different channels were installed less than one foot apart, horizontally. Vertically barriers and fire wraps were installed to separate cables and equipment and associated non-safety circuits of redundant trains to protect against radiant energy from a 10 CFR 50, Appendix R assumed fire.

Additionally, a horizontal barrier was installed where trays (other than heavy power) were installed less than one foot beneath any tray of a different channel.

In the area of the electrical tunnel between the Control Building and Containment Building and containment penetration area, two tunnels provide the separation for the four channels. A cross section of this portion of the tunnel is shown in the Plant Drawing 9321-F-31193 [Formerly Figure 7.2-18].

In general, control board switches with their associated indicating lights are contained in a modularized structure which provides physical separation between power "trains." Where more than one train is required to connect to a single switch, the wiring is routed to different quadrants within the module itself. Separate connectors for each redundant circuit are used, and board wiring is channelized to separate terminal blocks contained in individual channelized vertical risers located above separated floor slots. The wiring "trains" within the board are divided into three separate groups. Train "X" is that wiring which is associated with buses fed from diesel generator No. 32, Train "Y" is that wiring which is associated with buses fed from diesel generator No. 33 and Train "Z" is that wiring which is associated with buses fed from diesel generator No. 31. These "trains" are physically separated from each other by horizontal raceways which route the wiring to its appropriate vertical riser.

The wiring of local control panels which contain cabling from different channels have been separated by interior metal barriers or were separated into more than one panel. The main three phase power circuits are protected by means of three-pole breakers. Individual small power feeds from the motor control centers have three phase protection by means of fuses and "heater" overload devices. Single phase circuits are protected by single pole devices including fuses and/or breakers. (See Section 8.2)

Channel independence is carried throughout the system extending from the sensor to the relay actuating the protective function. The protective and control functions are fully isolated, control being derived from the primary protection signal path through an isolation amplifier. As such, a failure in the control circuitry does not affect the protection channel. This approach is used for pressurizer pressure and water level channels, steam generator water level, T_{avg} and ΔT channels, steam flow-feedwater flow and nuclear instrumentation channels.

The analog type equipment associated with the Reactor Protection and Engineered Safety Features Systems is considered to be the most susceptible to temperature effects because of the accuracies involved. Excessive temperature for long periods in areas containing switchgear, cables, etc. would result in a slight degradation of life but would not affect performance. The Control Room is the limiting case for reactor shutdown with regard to electrical equipment. The protective equipment in the control and relay rooms was designed to operate in an environment up to 120°F without loss of function.

Temperature in the Control Room and adjoining equipment room is maintained for personnel comfort at 70 ± 10°F. Protective equipment in this space was designed to operate within a design tolerance over this temperature range. Design specifications for this equipment specified no loss of protective function up to 120°F. Exceptions to this are evaluated in NSE 95-3-032, Revision 1 (See FSAR Section 9.9.2). Thus, there is a wide margin between design limits and the normal operating environment for control room equipment.

The engineered safety features equipment is actuated by one or the other of the engineered safety features actuation channels. Each coincidence network actuates an engineered safety actuation device that operates the associated engineered safety features equipment, motor starters and valve operators. As an example, the control circuit of a safety injection pump is typical of the control circuit for a large pump operated from switchgear. The actuation relay, energized by the Engineered Safety Features Instrumentation System, has normally open contracts. These contacts energize the circuit breaker closing coil to start the pump when the control relay is energized. The Engineered Safety Features Instrumentation System actuates (depending on the severity of the condition) the Safety Injection System, the Containment Isolation System, Containment Air Recirculation System and Containment Spray System.

In the Reactor Protection System, two reactor trip breakers are provided to interrupt power to the full length rod drive mechanisms. The breaker main contacts are connected in series (with power supply) so that opening either reactor trip breaker interrupts power to all full length rod mechanisms, permitting them to fall by gravity into the core.

In the event of a loss of reactor trip breaker control power, the reactor trip breaker under voltage coils and associated relays are de-energized and the breakers trip to an open mode. An electrical interlock prevents both bypass breakers from being closed concurrently.

Further detail on redundancy is provided through the detailed descriptions of the respective systems covered by the various sections in this chapter. In summary, reactor protection was designed to meet all presently defined reactor protection criteria. The original NRC Safety Evaluation Report evaluated the instrumentation and control systems against IEEE Standard 279-1968 and found the Reactor Protection System design to be acceptable.

Required continuous electrical supply is discussed in Chapter 8.

<u>Demonstration of Functional Operability of Protection Systems</u>

Criterion: Means shall be included for suitable testing of the active components of protection systems while the reactor is in operation to determine if failure or loss of redundancy has occurred. (GDC 25 of 7/11/67)

The analog equipment of each protection channel in service at power is capable of being tested and tripped independently by simulated analog input signals to verify its operation. The trip logic circuitry includes means to test each logic channel through to the trip breakers. Thus, the operability of each trip channel can be determined conveniently and without ambiguity.

Testing of the diesel-generator starting may be performed from the diesel generator control board. The generator breaker is not closed automatically after starting during this testing. The generator may be manually synchronized to the 480 Volt bus for loading. Complete testing of the starting of diesel generators can be accomplished by tripping the associated 480 Volt undervoltage relays and providing a coincident simulated safeguards signal. The ability of the units to start within the prescribed time and to carry load can be periodically checked. (The Electrical Systems are discussed in more detail in Section 8.2.3.)

The reactor coolant pump breakers open trip is not testable at power; it is a backup trip which is testable only during shutdown. Testing at power (opening the breakers) would involve a loss of flow in the associated loop.

<u>Protection Against Multiple Disability for Protection Systems</u>

Criterion: The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function or shall be tolerable on some basis. (GDC 23 of 7/11/67)

The components of the protection system were designed and laid out so that the mechanical and thermal environment accompanying any emergency situation in which the components are required to function does not interfere with that function.

Separation of redundant analog protection channels originates at the process sensors and continues back through the field wiring and containment penetrations to the analog protection racks. Physical separation is used to the maximum practical extent to achieve separation of redundant transmitters. Separation of field wiring is achieved using separate wire ways, cable trays, conduit runs and containment penetrations for each redundant channel. Redundant analog equipment is separated by locating redundant components in different protection racks. Each redundant channel is energized from a different vital instrument bus.

<u>Protection System Failure Analysis Design</u>

Criterion: The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electrical power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced. (GDC 26 of 7/11/67)

Each reactor trip circuit was designed so that trip occurs when the circuit is de-energized; therefore, loss of channel power causes the system to go into its trip mode. In a two-out-of-three circuit, the three channels are equipped with separate primary sensors and each channel is energized from an independent electrical bus. Failure to de-energize when required is a mode of malfunction that affects only one channel. The trip signal furnished by the two remaining channels is unimpaired in this event.

Reactor trip is implemented by interrupting power to the magnetic latch mechanisms on all drives allowing the full length rod clusters to insert by gravity. The protection system is thus inherently safe in the event of a loss of power.

The engineered safety features actuation circuits were designed on the “energize to operate” principle unlike the reactor trip circuits.

The steam line isolation signal on high-high containment pressure, which uses the same circuitry as the containment spray actuation signal, was also designed on the “energize to operate” principle. There are a total of six high-high containment pressure instruments which are separated into three channels. The three high-high containment pressure instrument channels are powered from three separate independent sources (one channel from instrument Bus No. 31 powered from Battery No. 31, the second channel from instrument Bus No. 33 powered from Battery No. 33, and the third channel from instrument Bus No. 34 powered from

Instrument Bus No. 34 powered from Battery No. 34 with alternate supply from safeguards Motor Control Center No. 36B).

This assures operation of a sufficient number of containment pressure instruments in the event of a power failure to one of the instrument channels.

In the event that power to any instrument bus is lost, there is no single failure that could occur to prevent any protective action. Reactor trip initiation signals are de-energized to actuate. The containment spray initiation signals, of which only two of three are required, are powered from three separate power sources (i.e., Instrument Buses No. 31, No. 33, and No. 34).

If power would ever be lost to any instrument bus, channel trip annunciators, etc. associated with the protective functions powered from this bus would alarm. This would mean to the operator that this one complete protective channel is in the trip mode. The event would be indicative of the loss of power for this particular channel of protective devices.

The above design is consistent with all of the instrument buses regardless of their source of power, as the loss of any one instrument bus, for any reason, would give channel trip alarms and indications for the respective channel of protection devices. These alarms would be a true indication because on loss of instrument power the associated protective channel is indeed in the trip mode. This complies with the requirements of Section 4.20 of IEEE-279. (See Section 8.2)

Each emergency diesel-generator is started by undervoltage on its associated 480 Volt bus or by the safety injection signal independent of the other 480 Volt buses and diesel generators. Engine cranking is accomplished by a stored energy system supplied solely for the associated diesel generators. The undervoltage relay scheme was designed so that loss of 480 Volt power does not prevent the relay scheme from functioning properly.

Redundancy of Reactivity Control

Criterion: Two independent control systems, preferably of different principles, shall be n provided. (GDC 27 of 7/11/67)

One of the two Reactivity Control Systems employs rod cluster control assemblies to regulate the position of Ag-In-Cd neutron absorbers within the reactor core. The other Reactivity Control System employs the Chemical and Volume Control System to regulate the concentration of boric acid solution (neutron absorber) in the Reactor Coolant System.

A detailed description of the Reactivity Control System for Indian Point 3, sufficient to demonstrate redundancy and capability as established under the provisions of this criterion, is presented in Section 3.1.

Reactivity Control Systems Malfunction

Criterion: The reactor protection system shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits. (GDC 31 of 7/11/67)

Reactor shutdown with rods is completely independent of the normal control functions since the trip breakers completely interrupt the power to the full length rod mechanisms regardless of existing control signals. Effects of continuous withdrawal of a rod control assembly and of deboration are described in Sections 7.3.1, 7.3.2, 9.2 and 14.1.

Principles of Design

Redundancy and Independence

The protective systems are redundant and independent for all vital inputs and functions. Each channel is functionally independent of other redundant channels and is supplied from an independent power source. Isolation of redundant protection channels is described in further detail elsewhere in this section and in Section 7.2.2.

Manual Actuation

Means are provided for manual initiation of protective system action. Failures in the automatic system do not prevent the manual actuation of protective functions. Manual actuation requires the operation of a minimum of equipment.

Channel Bypass or Removal from Operation

The system was designed to permit any one channel to be maintained and when required, tested or calibrated during power operation without system trip. During such operation the active parts of the system continue to meet the single failure criterion. Since the channel under test is either tripped or superimposed, test signals are used which do not negate the process signal.

It should be noted that the “one-out-of-two” logic systems are permitted to violate the single failure criterion during channel bypass, provided that acceptable reliability of operation can be otherwise demonstrated and bypass time interval is short.

Capability for Test and Calibration

The bistable portions of the protective system (e.g., relays, bistables, etc.) provide trip signals only after signals from analog portions of the system reach preset values.

Capability is provided for calibrating and testing the performance of the bistable portion of protective channels and various combinations of the logic networks during reactor operation.

The analog portion of a protective channel provides analog signals proportional to a reactor or plant parameter. The following means are provided to permit checking the analog portion of a protective channel during reactor operation:

a) Varying the monitored variable
b) Introducing and varying a substitute transmitter signal
c) Cross checking between identical channels or between channels which bear a known relationship to each other and which have readouts available.

The design permits the administrative control of the means for manually by-passing channels or protective functions.

The design permits the administrative control of access to all trip settings, module calibration adjustments, test points, and signal injection points.

Information Readout and Indication of Bypass

The protective systems were designed to provide the operator with accurate, complete, and timely information pertinent to their own status and to plant safety.

Indication is provided in the Control Room if some part of the system has been administratively bypassed or taken out of service.

Trips are indicated and identified down to the channel level.

Vital Protective Functions and Functional Requirements

The Reactor Protective System monitors parameters related to safe operation and trips the reactor to protect the reactor core against fuel rod cladding damage caused by departure from nucleate boiling (DNB) and to protect against Reactor Coolant System damage caused by high system pressure. The engineered safety features instrumentation system monitors parameters to detect failure of the Reactor Coolant System and initiates containment isolation and engineered safety features operation to contain radioactive fission products.

This section covers those protective systems provided to:

a) Trip the reactor to prevent or limit fission product release from the core and to limit energy release.
b) Isolate containment and activate the Isolation Valve Seal Water System when necessary.
c) Control the operation of engineered safety features provided to mitigate the effects of accidents.

The core protective systems in conjunction with inherent plant characteristics were designed to prevent anticipated abnormal conditions from causing fuel damage exceeding limits established in Chapter 3 or Reactor Coolant System damage exceeding effects established in Chapter 4.

Completion of Protective Action

Where operating requirements necessitate automatic or manual bypass of a protective function, the design is such that the bypass is removed automatically whenever permissive conditions are not met. Devices used to achieve automatic removal of the bypass of a protective function are part of the protective system and were designed in accordance with the criteria of this section.

The protective systems were designed so that once initiated, a protective action goes to completion. Return to normal operation requires administrative action by the operator.

Multiple Trip Settings

Where it is necessary to change to a more restrictive trip setting to provide adequate protection for a particular mode of operation or set of operating conditions, the design provides positive means of assuring that the more restrictive trip setting is used. The devices used to prevent

improper use of less restrictive trip settings are considered a part of the protective system and were designed in accordance with the other provisions of these criteria.

Interlocks and Administrative Procedures

Interlocks and administrative procedures required to limit the consequences of fault conditions other than those specified as limits for the protective function comply with the protective function comply with the protective system criteria.

Protective Actions

The Reactor Protective System automatically trips the reactor to protect the reactor core under the following conditions:

a) The reactor power, as measured by neutron flux, reaches a pre-set limit.
b) The temperature rise across the core, as determined from loop ΔT, reaches a limit either from an overpower ΔT set point or an overtemperature ΔT set point (function of T_{avg} and pressurizer pressure, adjusted by neutron flux distribution). Overtemperature ΔT set point is adjusted by neutron flux distribution.
c) The pressurizer pressure reaches an established minimum limit.
d) Loss of reactor coolant flow as sensed by low flow, loss of pump power or pump breakers opening.
e) Pressurizer pressure or level trips the reactor to protect the primary coolant boundary when the pressurizer pressure or level reaches an established maximum limit.

Interlocking functions derived from the Reactor Protective System inhibit control rod withdrawal on the occurrence of a specified parameter reaching a value lower than the value at which reactor trip is initiated.

For anticipated abnormal conditions, protective systems in conjunction with inherent plant characteristics and engineered safety features are designed to ensure that limits for energy release to the Containment and for radiation exposure (as in 10 CFR 100) are not exceeded.

Seismic Design Criteria

For either the operational or design basis earthquake, the equipment was designed to assure that it does not lose its capability to perform its function, i.e., shut the plant down and maintain it in a safe shutdown condition. For the design basis earthquake, permanent deformation of the equipment is acceptable provided that the capability to perform its function is maintained.

7.2.2 System Design

Reactor Protective System Description

Figure 7.2-2 is a block diagram of the Reactor Protective System; Figure 7.2-3 illustrates the core thermal limits and shows the trip points that are used for the protection system. The solid lines are a locus of limiting design conditions representing the core thermal limits at five pressures. The core thermal limits are based on the conditions which yield the applicable limit

value for departure from nucleate boiling ratio (DNBR) or those conditions which preclude bulk boiling at the vessel exit. The dashed lines indicate the maximum permissible trip points for the overtemperature high ΔT reactor trip including allowances for measurement and instrumentation errors.

The maximum and minimum pressures shown (2470 psia and 1750 psia) represent the set points for the high pressure and low pressure reactor trips.

Adequate margins exist between the worst steady state operating point (including all temperature, calorimetric, and pressure errors) and required trip points to preclude a spurious plant trip during design transients.

Indication

All transmitted signals (flow, pressure, temperature, etc.) which can cause a reactor trip are either indicated or recorded for every channel.

Engineered Safety Features Instrumentation Description

Plant Drawings IP3V-0171-0070, IP3V-0171-0056, 5651D72 Sheets 10, 12, and 12A [Formerly Figures 7.2-4, 7.2-5 and 7.2-6] show the action initiating sensors, bistables and logic for the engineered safety features instrumentation.

The engineered safety features actuation system automatically performs the following vital functions:

1) Start operation of the Safety Injection System upon low pressurizer pressure signal or high containment pressure signals (approximately 10% of containment design pressure), or on coincidence of high differential pressure between any two steam generators, 2 sets of 2/3 high-high pressure [energize to actuate], or after time delay (maximum 6 seconds) in coincidence with high steam flow in 2/4 lines in coincidence with (a) low T_{avg} in 2/4 lines or (b) low steam line pressure in 2/4 lines.

2) Operate the containment isolation valves in non-essential process lines upon detection of high containment pressure signals (Phase A containment isolation). The Isolation Valve Seal Water System is actuated upon automatic actuation of the Safety Injection System.

3) Start the Containment Spray System and operate the remaining containment isolation valves upon detection of a containment pressure signal higher than required in item (2) above (Phase B containment isolation; approximately 24 psig).

4) Start operation of the safeguards equipment actuation sequence signal. This includes actuating signals to such components as the Safety Injection System and the Containment Air Recirculation, Cooling and Filtration System.

Steam Line Isolation

Any of the following signals will close all steam line isolation valves:

1) After time delay (maximum 6 seconds) in coincidence with high steam flow in 2/4 lines in coincidence with (a) low T_{avg} in 2/4 lines or (b) low steam line pressure in 2/4 lines.

2) High containment pressure signals (two sets of 2/3 high-high pressure) [energize to actuate].
3) Steam line isolation valves can also be closed one at a time by manual action.

Feedwater Line Isolation

Any safety injection signal will isolate the main feedwater lines by closing all control valves (including associated MOVs) and the pump discharge valves. The closure of the pump discharge valves will cause the main feedwater pumps to trip.

ATWS Mitigating System Actuation Circuitry (AMSAC) Description

The ATWS Mitigating System Actuation Circuitry (AMSAC) is installed at IP3 in accordance with the requirements of 10 CFR 50.62 “Reduction of Risk From Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants.” An ATWS is an anticipated operational occurrence (such as loss of feedwater, loss of condenser vacuum, or loss of offsite power) that is accompanied by a failure of the Reactor Protection System (RPS) to shut down the reactor. The ATWS Rule requires specific improvements in the design and operation of commercial nuclear power facilities to reduce the probability of failure to shut down the reactor following anticipated transients and to mitigate the consequences of an ATWS event.

AMSAC provides an alternate means of tripping the turbine and actuating auxiliary feedwater (AFW) flow apart from the reactor protection system (RPS). The AMSAC equipment is reasonably diverse from the existing RPS equipment to minimize the potential for common cause failures. Also, AMSAC logic power supplies and logic circuitry are independent from the RPS power supplies and logic circuitry. The turbine trip and AFW flow actuation will provide adequate assurance that the reactor coolant system (RCS) would not be subject to potential damage as a result of overpressure. The pressure limit (3200 psig) corresponds to the ASME Boiler and Pressure Vessel Code Level C Service Limit stress criteria. Past ATWS analyses, see WCAP-8330 for example, show there are only two ATWS transients for which the ASME Service level limit may be approached. These transients are the Complete Loss of Normal Feedwater Without Scram and the Loss of Load Without Scram.

The Complete Loss of Normal Feedwater transient can occur due to the simultaneous tripping of the main feedwater or condensate pumps or the simultaneous closing of the main feedwater control valves or main feedwater pump discharge valves.

The Loss of Load transient considered for ATWS is one in which the vacuum in the main condenser is lost, resulting in a complete loss of normal feedwater. This could occur, for example, if the circulating water pumps trip. The main turbine will then trip on high backpressure as will any turbine-driven main feedwater pump that exhausts into the main condenser.

Since, in both of the above described transients (and in only these transients) the main feedwater is completely lost, the AMSAC is designed to actuate the auxiliary feedwater flow when the complete loss of main feedwater flow is anticipated.

Short-term protection against high reactor coolant system pressures is not required until 70% of nominal power. However, in order to minimize the amount of reactor coolant system voiding during an ATWS, AMSAC operates at and above 40% of turbine power. Furthermore, the

potential exists for spurious AMSAC actuations during start-up at the lower power levels. To assure the above requirements are met, AMSAC is automatically blocked at turbine loads less than 40% by the C-20 permissive. In the event of a turbine trip, both turbine power transmitter indications will drop below 40% of full scale turbine power level. A timer in the AMSAC circuitry will maintain the trip permissive (C-20) for 330 seconds to ensure that the AMSAC system is still armed. However, in the event of an ATWS below 40% of nominal load, operator action will be required to provide long-term core protection by initiating auxiliary feedwater flow.

Actuation of AMSAC will occur on low main feedwater flow as measured by the low feedwater flow transmitters. The setpoint to actuate AMSAC is approximately 21% of nominal main feedwater flow. Although 21% flow is more than ample to protect against overpressure in the event of an ATWS, instrumentation error would become unacceptably large if a substantially lower set point were used.

An AMSAC output is initiated after a predetermined time delay whenever turbine power is 40% or greater coincident with three of the four feedwater flow transmitters indicating feedwater flow of 21% or less. The time delay is determined by the highest Turbine Power Level sensed at the time the ¾ low feedwater flow is sensed. 60 second lag units maintain Turbine Power Level close to the pre-turbine trip condition, for determination of the variable time delay. The time delay varies from a maximum of 300 seconds at 40% power to 25 seconds at 100% power (in accordance with the WOG curves). The purpose of this time delay is twofold. First, this time delay allows the reactor protection system to respond initially to a low feedwater flow condition. Secondly, during this time delay, the operator is provided with an AMSAC alert annunciator in the CR. If during the AMSAC alert period the operator increases feedwater flow above 21%, AMSAC will not actuate and the timer will reset. However, once an AMSAC signal is initiated, the signal will be maintained for at least 40 seconds to ensure all required actions occur. Turbine trip, turbine power auxiliary feedwater valve actuation and steam generator isolation and sample valve closure functions are immediately actuated by AMSAC. The motor driven auxiliary feedwater pumps have a 28 second time delay built into their starting circuits. As such, the motor driven auxiliary feedwater pumps will start 28 seconds after an AMSAC signal is initiated. This time delay is in accordance with 10 CFR 50.62 (the AMSAC Rule) which requires that the AMSAC AFW initiation function is performed within 90 seconds following initiation of an AMSAC signal. The AMSAC output signal is energized to actuate, so that a loss of power to the AMSAC cabinet will not initiate an AMSAC trip.

The AMSAC Logic Diagram is shown in Plant Drawing 9321-LL-38077 [Formerly Figure 7.2-19].

Reactor Protective System Safety Features

Separation of Redundant Protection Channels

The Reactor Protection System was designed on a channelized basis to achieve separation between redundant protection channels. The channelized design, as applied to the analog as well as the logic portions of the protection system, is illustrated by Figure 7.2-1 and is discussed below. Although shown for four channel redundancy, the design is applicable to two and three channel redundancy.

Separation of redundant analog channels originates at the process sensors and continues through the field wiring and containment penetrations to the analog protection racks.

Physical separation was used to the maximum practical extent to achieve separation of redundant transmitters. Separation of field wiring was achieved using separate wireways, cable trays, conduit runs and containment penetrations for each redundant channel. Analog equipment was separated by locating redundant components in different protection racks. Each redundant protection set is energized from a separate AC power feed.

The reactor trip bistables are mounted in the protection racks and are the final operational component in an analog protection channel. Each bistable drives two logic relays ("C" & "D"). The contacts from the "C" relays are interconnected to form the required actuation logic for Trip Breaker No. 1 through DC power feed No. 1. The transition from channel identity to logic identity is made at the logic relay coil/relay contact interface. As such, there is both electrical and physical separation between the analog and the logic portions of the protection system. The above logic network is duplicated for Trip Breaker No. 2 using DC power feed No. 2 and the contacts from the "D" relays. Therefore, the two redundant reactor trip logic channels will be physically separated and electrically isolated from one another. Overall, the protection system is comprised of identifiable channels which are physically, electrically and functionally separated and isolated from one another.

Physical Separation

The physical arrangement of all elements associated with the protective system reduces the probability of a single physical event impairing the vital functions of the system.

System equipment is distributed between instrument cabinets so as to reduce the probability of damage to the total systems by some single event.

Wiring between vital elements of the system outside of equipment housing was routed and protected so as to maintain the true redundancy of the systems with respect to physical hazards. The same channel isolation and separation criteria as described for the reactor protection circuits were applied to the engineered safety features actuation circuits.

Loss Power

A loss of power in the Reactor Protective System causes the affected channel to trip. All bistables operate in a normally energized state and go to a de-energized state to initiate action. Loss of power, thus, automatically forces the bistables into the tripped state.

Availability of power to the engineered safety features instrumentation is continuously indicated. The loss of instrument power to the sensors in the engineered safety feature instrumentation starts the engineered safety features equipment associated with the affected channels, except for containment spray which requires instrument power for actuation. Steam line isolation on high-high containment pressure, which utilizes the same actuation circuitry as the containment spray actuation, also requires power to actuate. There are a total of six high-high containment pressure instruments which are separated into three instrument channels. The three high-high containment pressure instrument channels are powered form three separate, independent sources to assure operation in the event of a power failure to one of the instrument channels.

Engineered Safety Features Systems Testing

At least once per 24 months, the master relays will be operated with test input to actuate the safeguards sequences. The test will be terminated upon verification that the associated valves

are properly aligned and associated pumps are started by the automatic actuation circuits. No flow is introduced into the Reactor Coolant System; verification of pump startup is by breaker position indication and visual inspection of local flow meters in the mini-flow lines, where applicable. The tests will be performed in accordance with the Technical Specification.

Process Analog Protection Channel Testing

The basic arrangement of elements comprising a representative analog protection channel is shown in Figure 7.2-7. These elements include a sensor or transmitter, power supply, bistable, bistable trip switch and proving lamp, test-operate switch, test annunciator, test signal injection jack, and test points. A portion of the logic system is also included to illustrate the overlap between the typical analog channel and the corresponding logic circuits. The analog system symbols are given in Figure 7.2-14.

Each protection rack include a test panel containing those switches, test jacks and related equipment needed to test the channels contained in the rack. An interlocked hinged cover encloses the test panel. Opening the cover or placing the test-operate switch in the “TEST” position automatically initiates an alarm. These alarms are arranged in rack “sets” to annunciate entry to more than one rack or redundant protection “sets” or channels at any time. The test panel cover is designed such that it cannot be closed (and the alarm cleared) unless the test signal plugs (described below) are removed. Closing the test panel cover mechanically returns the test switches to the “OPERATE” position.

Test procedures allow the bistable output relays of the channel under test to be placed in the tripped mode prior to proceeding with the analog channel tests. Thus, for the channel under test, the relay elements in the two-out-of-three or the two-out-of-four coincident matrices will be in the tripped mode during the entire test of the channel. This ensures that the remaining channels of the two-out-of-three or the two-out-of-four protective functions meet the single failure criterion during the entire channel test. Placing the bistable trip switch in the tripped mode de-energizes (trips) the bistable output relays and connects a proving lamp to the bistable output circuit. This permits the electrical operation of the solid-state bistable to be observed and the bistable set point relative to the channel analog signal to be verified. Test procedures also allow the bistable output relays of the channel under test to be placed in the bypassed mode prior to proceeding with the analog channel test; i.e., a two-out-of-three circuit becomes a two-out-of-two circuit. Testing in bypass mode is depicted in Figures 7.2-20, 7.2-21, and 7.2-22. This may only be done for circuits whose hardware does not require the use of jumpers or lifted leads to be placed in the bypass mode. Upon completion of test of the analog channel, the bistable trip switches must be manually reset to their operate mode. Closing the cover of the test panel will not transfer the bistable trip switches from their tripped to their operate position.

The following circuits are equipped with trip bypass capability:

REACTOR TRIP	**AUTO SAFETY INJECTION ACTUATION**
Overpower Delta T	Hi Containment Pressure
Over Temperature Delta T	Steam Line Delta P
Lo Steam Generator Level	Hi Steam Flow SI
Lo-Lo Steam Generator Level	Lo Steam Line Pressure
Steam Flow > Feedwater Flow Mismatch	Lo Tavg
Pressurizer Hi Pressure	Lo Pressurizer Pressure

Pressurizer Lo Pressure	
Pressurizer Hi Level	**TURBINE TRIP**
Lo Reactor Coolant Flow	Steam Generator Hi-Hi Level
Stop Rod Withdrawal	

Analog channel tests are accomplished by simulating a process measurement signal, varying the simulated signal over the signal span and checking the correlation of bistable set points, channel readouts and other loop elements with precision portable read-out equipment. Test jacks are provided in the test panel for injection of the simulated process signal into each process analog protection channel. Test points are provided in the channel to facilitate an independent means for precision measurement and correlation of the test signal. This procedure does not require any tools nor does it involve in any way the removal or disconnection of wires in the channel under test. In general, the analog channel circuits are arranged so that the channel power supply is loaded and is providing sensing circuit power during channel test. Load capability of the channel power supply is thereby verified by the channel test.

Nuclear Instrumentation Channel Testing

Nuclear Instrumentation Channel Systems (NIS) channels are tested by superimposing the test signal on the actual detector signal being received by the channel. The output of the bistable is not placed in a tripped condition prior to testing. A valid trip signal would then be added to the existing test signal, and thereby cause channel trip at a somewhat lower percent of actual reactor power. Protection bistable operation is tested by increasing the test signal (level signal) to the bistable trip level and verifying operation at control board alarms and/or at the NIS racks.

A NIS channel which can cause a reactor trip through one-out-of-two protection logic (source or intermediate range) is provided with a bypass function which prevents the initiation of a reactor trip from that particular channel during the short period that it is undergoing test. The power range channels do not require bypass of the reactor trip function for test purposes since the protection logic is two-out-of-four. The power range dropped rod function is operated from a one-out-of-four protection logic; therefore, a bypass function is provided on each of the power range channels to prevent load cutback during the dropped rod channel test. Over-riding the dropped rod circuitry from causing a spurious turbine runback due to instrument bus noise has no impact on the utilization of the Rod Drop Bypass Switch on each Power Range Nuclear Instrument for nuclear instrument testing.

In all cases the bypass condition and the channel test condition are alarmed on the NIS drawer and at the main control board. An interlock feature between the bypass switch and channel test switch on each channel keeps the test signal from being activated until the bypass function has been inserted. Administrative control is required to ensure that only one protection channel is placed in the bypass condition at any one time. The power range reactor trips are not affected by the bypass function described above. Therefore these power range trips will be active if required. No provision was made in the channel test circuit for reducing the channel signal level below that signal being received from the NIS detector.

Logic Channel Testing

The general design features of the logic system are described below. The trip logic channels for a typical two-out-of-three and a two-out-of-four trip function are shown in Figure 7.2-8. The

analog portions of these channels are shown in Figure 7.2-9. Each bistable drives two relays ("A & B" for level and "C" & "D" for pressure). Contacts from the "A" and "C" relays are arranged in a 2/3 and 2/4 trip matrix for Trip Breaker No. 1 (RTB). The above configuration is duplicated for Trip Breaker No 2 (RTA) using contacts from the "B" and "D" relays. A series configuration is used for the trip breakers since they are actuated (opened) by undervoltage coils. This approach is consistent with a de-energize-to-trip preferred failure mode. The planned logic system testing includes exercising the reactor trip breakers to demonstrate system integrity. Bypass breakers are provided for this purpose. During normal operation, these bypass breakers are open. Administrative control is used to minimize the amount of time these breakers are closed. Closure of the breaker is controlled from its respective logic test panel in the Control Room. An interlock is provided that trips both bypass breakers open if a second bypass breaker is closed. The status of the breaker is indicated in the Control Room by indicating lights.

As shown in Figure 7.2-8 the trip signal from the logic network is simultaneously applied to the main trip breaker associated with the specific logic chain as well as the Bypass Breaker associated with the alternate trip breaker. Should a valid trip signal occur while Bypass Breaker No. 1 (BYB) is bypassing Trip Breaker No. 1 (RTB), Trip Breaker No. 2 (RTA) will be opened through its associated logic train. The trip signal applied to Trip Breaker No. 2 (RTA) is simultaneously applied to bypass breaker No. 1 (BYB) thereby opening the bypass around Trip Breaker No. 1 (RTB). RTB would either be opened manually as part of the test or would be opened through its associated logic train which would be operational or tripped during a test. Two auxiliary relays are located in parallel with the undervoltage coils of the trip breaker. The output contacts (normally closed) of these relays are connected in series and initiate actuation of the shunt trip coil of both the reactor trip and the associated bypass breaker upon a reactor trip signal. The above contacts are connected to the respective breaker shunt trip coil circuit through test switches which, during the testing of the undervoltage trip device, block the undervoltage trip signal. The test switches are supervised by control room annunciation. In addition, key operated test switches are provided for each train to allow energization of breaker shunt trip coil independent of the undervoltage trip device. The two sets of test switches in conjunction permits selection of particular reactor or bypass breaker to be tested. During response time testing, the shunt trip relay is tied to a portable recorder which is used to indicate transmission of a trip signal through the logic network. Lights are also provided to indicate the status of the individual logic relays.

The following procedure illustrates the method used for testing Trip Breaker No. 1 (RTB) and its associated logic network:

a) Manually set and trip Bypass Breaker No. 1 (BYB) to verify operation.
b) Set BYB; trip Trip Breaker No. 1 (RTB).
c) Place key operated switch "Train-Auto Defeat" to test position, verify alarm and test lamp illumination.
d) Sequentially de-energize the trip relays 9A1, A2, A3) for each logic combination (1-2, 1-3, 2-3). Verify that the logic network de-energizes the UV coil on Trip Breaker No. 1 (RTB) for each logic combination. Since the neon light monitors the signal applied to the UV coil, operation of the UV coil can be determined from the neon light.
e) Repeat "D" for every logic combination in each matrix.
f) Reset Trip Breaker No. 1 (RTB).
g) Trip RTB to validate prior test results as evidenced by the neon light.
h) Reset Trip Breaker No. 1 (RTB). Trip BYB.

In order to minimize the possibility of operational errors from either the standpoint of tripping the reactor inadvertently or only partially checking all logic combinations, each logic network includes a logic channel test panel. This panel includes those switches, indicators and recorders needed to perform the logic system test. The front panel arrangement is shown in Figure 7.2-10. The test switches used to de-energize the trip bistable relays operate through interposing relays as shown in Figures 7.2-7 and 7.2-9. This approach avoids violating the separation philosophy used in the analog channel design. Thus, although test switches for redundant channels are conveniently grouped on a single panel to facilitate testing, physical and electrical isolation of redundant protection channels are maintained by the inclusion of the interposing relay which is actuated by the logic test switches.

If the logic test switches in both engineered safeguards logic trains are placed in the test mode simultaneously, the automatic safeguards actuation will be blocked for the two trains. However, a separate alarm on the main control board is provided for each safeguard train to indicate when each train is in test.

The test switches are located in separate safeguards racks and administrative control prevents the simultaneous operation of Train A and Train B test switches.

It should be noted that either one of the safeguards train, which is blocked by its test switch, can always be unblocked and actuated by the manual safety injection switch at the main control board.

Safeguards Initiating Circuitry

The safeguards actuation circuitry and hardware layout are designed to maintain circuit isolation through the bistable operated logic relays. The channelized design follow through is shown on the Figure 7.2-15 block diagram.

The orderly arrangement of equipment for the Reactor Protection System and Engineered Safety Features Actuation System helps facilitate testing and maintenance. A color code of red, white, blue and yellow is used for analog protection channels in sets I, II, III, and IV, respectively. Large identification plates with the appropriate background color are attached at the front and back surfaces of each analog rack. The protection logic cabinets, housing the Train A logic, master relays, and slave relays, are physically separated from cabinets housing Train B equipment and identified by large identification plates on the input side of the racks where protection signals from the various protection channels are received. Small electrical components have nameplates on the enclosure which houses them. All cables are numbered with identification tags. These numbers are cross-referenced with cable schedule which specifies cable routing and function. The cable trays are color coded with each of the four channels having a different color assigned.

The safeguards bistables, mounted in the analog protection racks, drive both “A” and “B” logic matrix relays. Each matrix contains its own test light and test circuitry. The “A” and “B” logic matrices operate master relays for actuating channels A and B respectively, as shown in Figure 7.2-16.

Control power for logic channels A and B, is supplied from DC distribution panel No. 31 and No. 34, respectively. These redundant actuating channels operate the various safeguards components required with the large loads sequenced as necessary.

Protection channel identity is lost in the intermixing of the relay matrix wiring. Separation of A and B logic channels is maintained by the separate logic racks.

For safety injection, manual reset of the safeguards actuation relays may be accomplished two minutes following their operation. Once reset action is taken, the master relay is reset and its operation blocked, except for manual initiation. The engineered safeguards circuitry can be unblocked by resetting the reactor trip breaker.

Hinged safety covers on the reset pushbuttons in the circuitry of the Safety Injection, Containment Spray, Containment Isolation Phase A and Phase B, and Containment Ventilation Isolation Systems require deliberate action by the operators to actuate these pushbuttons and facilitate placing adequate administrative controls on the actuation of these pushbuttons. The Containment Ventilation Isolation System cannot be placed in a bypass condition while any of the automatic safety signals is present.

Separate and independent key-lock switches, one for each SI train, are provided in series to each of the auto SI actuation relays to allow manual blocking of the Engineered Safeguards System actuation. (See Section 6.2.2)

Logic Channel Testing

Figures 7.2-16 and 7.2-17 show the basic logic test scheme. Test switches are located in associated relay racks rather than in a single test panel. The following procedure is used for testing the logic matrices:

1) Following administrative procedure, test Channel A or B, one at a time

2) Depress the test relay switch to energize the rack test relays. An alarm will sound on the main board and a light at the rack will indicate that the safeguards rack is now in test.

3) Select a matrix and depress the logic test switches. The master relay will energize and matrix test lights will indicate upon actuation of the particular matrix being tested. The slave relay test lights will verify that the master relay contact associated with a particular slave relay has functioned and will also verify the integrity of the slave relay coils.

4) Reset the master relay by depressing the master relay reset switch. Reset the test relays by depressing the test reset switch. A lamp will glow as long as the test relays are energized. If a test relay contact in a particular slave relay circuit does not return to its normal position, then the slave relay test lamps will indicate such. Test lights can be tested by depressing the lens.

Primary Power Source

The primary source of control power for the Reactor Protective System is the vital instrument buses described in Chapter 8. The source of power for the measuring elements and the actuation circuits in the engineered safety features instrumentation is also from those buses.

Protective Actions

Reactor Trip Description

The Reactor Protection System acts to shut the reactor down by means of various reactor trips which are designed to occur when a measured plant variable exceeds predetermined limits. The protection system consists of all instrumentation which monitors the process variables and initiates trip if the process variables approach safety limits. It includes, but is not limited to, sensing elements, transmitters, converters, relays, actuating devices, interlocks, alarms, signal lines, etc. The trips function to provide rapid reduction of reactivity by the insertion of full-length RCC assemblies under free fall into the reactor core. The full-length RCC assemblies must be energized to remain withdrawn from the core.

Automatic reactor trip occurs upon the loss of power to the full-length control rods. All power to the full-length control rod mechanisms are interlocked by duplicate series connected circuit breakers. The trip breakers are opened by the undervoltage coils on both breakers. The undervoltage coils, which are normally energized, become de-energized by any one of the several trip signals.

Certain reactor trip channels (low reactor coolant flow, etc.) are automatically bypassed at low power where they are not required for safety. Nuclear source range, intermediate range and power range (low setpoint) trips, which are specifically provided for protection at low power or subcritical operation, are bypassed by operator manual action after receiving a permissive signal from the next higher range of instrumentation to allow power escalation during startup.

During power operation, a sufficiently rapid shutdown capability in the form of RCC assemblies is administratively maintained through the control rod insertion limit monitors. Administrative control requires that all shutdown rods be in the fully withdrawn position during power operation.

A resume of reactor trips, including means of actuation and the coincident circuit requirements, is given in Table 7.2.1. The permissive circuits referred to (e.g., P-7) are listed in Table 7.2-2.

Manual Trip

The manual actuating devices are independent of the automatic trip circuitry and are not subject to failures which might make the automatic circuitry inoperable. Either of two manual trip devices located in the Control Room will initiate a reactor trip.

High Nuclear Flux (Power Range) Trip

This circuit trips the reactor when two of the four power range channels read above the trip setpoint. There are two independent trip settings, one high and one low setting. The high trip setting provides protection during normal power operation. The low setting, which provides protection during startup, can be manually bypassed when two out of the four power range channels read above approximately 10% power (P-10). Three out of the four channels below 10% automatically reinstates the trip protection. The high setting is always active.

High Nuclear Flux (Intermediate Range)Trip

This circuit trips the reactor when one out of the two intermediate range channels reads above the trip setpoint. This trip, which provides protection during reactor startup, can be manually bypassed if two out of four power range channels are above approximately 10% (P-10). Three out of four channels below this value automatically reinstate the trip protection. The intermediate channels (including detectors) are separate from the power range channels.

High Nuclear Flux (Source Range) Trip

This circuit trips the reactor when one of the two source range channels reads above the trip setpoint. The trip, which provides protection during reactor startup, can be manually bypassed when one of two intermediate range channels reads above the P-6 setpoint value and is automatically reinstated when both intermediate range channels decrease below this value (P-6). This trip is also bypassed by two out of four high power range signals (P-10). It can also be reinstated below P-10 by an administrative action requiring coincident manual actuation.

The trip point is set between the intermediate range lower limit of instrument sensitivity and the upper limit of the source range instrument range.

Overtemperature ΔT Trip

The purpose of this trip is to protect the core against DNB. This circuit trips the reactor on coincidence of two-out-of-the-four signals with one channel (two temperature measurement, hot and cold) per loop. The set point for this reactor trip is continuously calculated for each channel by solving equations of this form:

$$\Delta T_{trip} - \Delta T_o [K_1 - K_2 (T_{avg} - T') + K_3 (P - P') - f(\Delta I)]$$

where

ΔT_o	-	indicated ΔT at rated power, F
T_{avg}	-	reactor coolant average temperature, two measurements in each loop (T_{avg} signal is rate compensated), F
T'	-	indicated T_{avg} at nominal condition at rated power, F
P	-	pressurizer pressure, four independent measurements, psia
P'	-	nominal pressure at rated power, psia
K_1	-	set point bias, F
K_2, K_3	-	constants based on the effect of temperature and pressure on the DNB limits
f (ΔI)	-	a function of the indicated difference between top and bottom detectors of the power range nuclear ion chambers with gains selected based on measured instrument response during plant startup tests.

Overpower ΔT Trip

The purpose of this trip is to protect against excessive power (fuel rod rating protection). This circuit trips the reactor on coincidence of two out of the four signals with one channel (one pair of temperature measurements) per loop.

The set point for this reactor trip is continuously calculated for each channel by solving equations of the form;

$$\Delta T_{set\ point} - \Delta T_o [K_4 - K_5 \frac{dT_{avg}}{dt} - K_6 (T_{avg} - T')]$$

where

ΔT_o	-	indicated ΔT at rated power, F
T_{avg}	-	Average temperature, F
T'	-	Indicated T_{avg} at nominal conditions at rated power, F
K_4	-	Set point bias
K_5	-	Constant
K_6	-	Constant

Low Pressurizer Pressure Trip

The purpose of this circuit is to protect against excessive core steam voids which could lead to DNB. The circuit trips the reactor on coincidence of two out of the four low pressurizer pressure signals. This trip is blocked when any three of the four power range channels and two of two turbine first stage (inlet) pressure channels read below approximately 10% power (P-7).

High Pressurizer Pressure Trip

The purpose of this circuit is to limit the range of required protection from the overtemperature ΔT trip and to protect against Reactor Coolant System over-pressure. This circuit trips the reactor on coincidence of two out of the three high pressurizer pressure signals.

High Pressurizer Water Level Trip

This trip is provided as a backup to the high pressurizer pressure trip. The coincidence of two out of the three high pressurizer water level signals trips the reactor. The trip is bypassed when any three of the four power range channels and two of the two turbine first stage (inlet) pressure channels read below approximately 10% power (P-7).

Low Reactor Coolant Flow Trip

The trip protects the core form DNB following a loss of coolant flow accident. The means of actuating the loss of coolant flow accident trip are:

a) Measured low flow in the reactor coolant loop. The low flow trip signal is actuated by the coincidence of 2/3 signals of any reactor coolant loop. The loss of flow in any two loops causes a reactor trip above approximately 10% power (P-7). Above the P-8 setpoint any

one loop causes a reactor trip. The sensor used for flow measurement is an elbow tap and is discussed in Chapter 4.

b) Reactor coolant pump circuit breaker open functions similarly to the low flow signal with one sensor per reactor coolant pump breaker.

c) Underfrequency on any two of the four reactor coolant pump buses will trip all four reactor coolant pumps and cause a reactor trip above approximately 10% power (P-7).

d) Undervoltage on any two of the four reactor coolant pump buses causes a direct reactor trip above approximately 10% power (P-7).

Safety Injection System (SIS) Actuation Trip

A reactor trip occurs when the Safety Injection System is actuated. The means of actuating the SIS trips are:

1) Low pressurizer pressure (two out of three). This signal may be manually blocked or unblocked during start-up and shutdown. This block is accomplished by separate switches for each of the redundant safety injection initiation circuits. The block will be automatically removed above a designated setpoint.

2) High containment pressure (two out of three) set at approximately 10% of containment design pressure.

3) High differential pressure between any two steam lines (two out of three).

4) After time delay: high steam flow in 2/4 lines (one out of two per line), in coincidence with either low T_{avg} in 2/4 lines or low steam line pressure in 2/4 lines.

5) High-high containment pressure (two sets of two-out-of-three), set at approximately 50% of containment design pressure [energize to actuate].

6) Manual.

Turbine Generator Trip

A turbine trip is sensed by two out of three signals from auto-stop oil pressure. A turbine trip is accompanied by a direct reactor trip above P-8 and a controlled short term release of steam to the condenser occurs which removes sensible heat from the Reactor Coolant System while avoiding steam generator safety valve actuation. Any reactor trip will generate a turbine trip. Further details are discussed in Chapter 10.

Steam/Feedwater Flow Mismatch Trip

This trip protects the reactor from a sudden loss of heat sink. The trip is actuated by one steam/feedwater flow mismatch in selected coincidence with one low steam generator water level in that steam generator. There are two steam/feedwater flow mismatches and two low steam generator water level signals per loop.

Low-Low Steam Generator Water Level Trip

The purpose of this trip is to protect the steam generators for the case of a sustained steam/feedwater flow mismatch. The trip is actuated on two out of the three low-low water level signals in any steam generator. A diagram of the steam generator level control and protection system is shown in Plant Drawing IP3V-0171-0355 [Formerly Figure 7.2-13].

Rod Stops

A list of rod stops is listed in Table 7.2-3. Some of these have been previously noted under permissive circuits, but are listed again for completeness.

Rod Drop Protection

Two independent systems are provided to sense a dropped rod: a rod bottom position detection system and a system which senses sudden reduction in out-of-core neutron flux. Both protection systems initiate protective action in the form of blocking automatic rod withdrawal [Deleted].

The primary protection for the dropped RCC accident is the rod bottom signal derived for each rod from its individual position indication system. With the position indication system, initiation of protection is independent of rod location of reactivity worth.

Backup protection is provided by use of the out-of-core power range nuclear detectors and is particularly effective for large nuclear flux reductions occurring in the region of the core adjacent to the detectors.

The rod drop detection circuit from nuclear flux consists basically of a comparison of each ion chamber signal with the same signal taken through a first order lag network. Since a dropped RCC assembly will rapidly depress the local neutron flux, the decrease in flux will be detected by one or more of these four sensors. Such a sudden decrease in ion chamber current will be seen as a difference signal. A negative signal output greater than a preset value (approximately 10%) from any of the four power range channels will actuate the rod drop protection.

Figure 7.4-2 indicates schematically the dropped rod detection circuits and the Nuclear Protection System in general. The potential consequences of any dropped RCC without protective action are presented in Section 14.1.4.

Alarms

Any of the following conditions actuate an alarm:

a) Reactor trip (first-out annunciator)
b) Trip of any reactor trip channel
c) Significant deviation of any major control variable (pressure, T_{avg}, pressurizer water level, and steam generator water level)
d) Actuation of any permissive circuit or override. (Certain permissive are provided with indication light only on the flight panel.)

Control Group Rod Insertion Limits

The control rod insertion limit system is used in an administrative control procedure with the objective to maintain an RCCA shutdown margin.

The control group rod insertion limits, Z_{LL}, are calculated as a linear function of reactor power and reactor coolant average temperature. The equation is:

$$Z_{LL} - A\,(\Delta T)_{avg} + B\,(\overline{T_{avg}}) + C$$

where A and B are preset manually adjustable gains and C is a preset manually adjustable bias. These set points may be different for each control bank. The $(\Delta T)_{avg}$ and $(\overline{T_{avg}})$ are the average of the individual temperature differences and the coolant average temperatures, respectively, measured from the reactor coolant hot leg and cold leg.

One insertion limit monitor with two alarm set points is provided for each control bank. A description of control and shutdown rod groups is provided in Section 7.3. The low alarm alerts the operator of an approach to a reduced shutdown reactivity situation requiring boron addition by following normal procedures with the Chemical and Volume Control System (Chapter 9). Actuation of low-low alarm requires the operator to take immediate action to add boron to the system by any one of several alternate methods.

7.2.3 System Evaluation

Reactor Protection System and DNB

The following is a description of how the reactor protection system prevents DNB.

The plant variables affecting the DNB ratio are:
- Thermal power
- Coolant flow
- Coolant temperature
- Coolant pressure
- Distribution Core power (hot channel factors)

Figure 7.2-11 illustrates the core limits for which DNBR for the hottest rod is at the design limit and shows the overpower and overtemperature ΔT reactor trips locus as a function of T_{avg} and pressure.

Excessive axial offset reduces the overtemperature ΔT setpoint associated with both the block on control rod withdrawal and the reactor trip actuation. If the ΔT of any RCS loop exceeds the calculated overpower or overtemperature ΔT setpoints, permissive signals will be generated which will initiate a block on control rod withdrawal. The setpoint on these ΔT rod blocks are approximately 2^{o} F less than the corresponding ΔT setpoints used to actuate reactor trip. [Deleted] Rod block on ΔT circuitry is not redundant, whereas the ΔT reactor trips are protective grade and meet the standards of IEEE-279.

Reactor trips for a fixed high pressurizer pressure and for a fixed low pressurizer pressure are provided to limit the pressure range over which core protection depends on the variable overpower and overtemperature ΔT trips.

Reactor trips on nuclear overpower and low reactor coolant flow are provided for direct, immediate protection against rapid changes in these variables. However, for all cases in which the calculated DNBR approaches the applicable DNBR limit, a reactor trip on overpower and/or overtemperature ΔT would be actuated.

The ΔT trip functions are based on the differences between measurements of the hot leg and cold leg temperatures, which are proportional to core power.

The overtemperature ΔT trip function is provided with a nuclear flux feedback to reflect a measure of axial power distribution. This will assist in preventing an adverse distribution which could lead to exceeding allowable core conditions.

Overpower Protection

In addition to the high power range nuclear flux trips, an overpower ΔT trip is provided (2 out of 4 logic) to limit the maximum overpower.

A rod stop function [Deleted] is provided in the form:
ΔT rod stop = ΔT trip - B_p
B_P = set point bias (F)

The logic for the rod stop is one out of four.

Overtemperature Protection

A second ΔT trip (2 out of 4 logic) provides an overtemperature trip which is a function of coolant average temperature and pressurizer pressure derived as previously discussed.

A similar rod stop function is provided in the form;
ΔT rod stop = ΔT trip - B_T
B_T = set point bias, (F)

The logic for the rod stop is one out of four.

In summary, in the event the difference between top and bottom detectors exceeds the desired range, automatic feedback signals are provided to reduce the overtemperature trip setpoint and to block rod withdrawal to maintain appropriate operating margins to the trip setpoint.

Interaction of Control and Protection

The design basis for the control and protection systems permits the use of a detector for both protection and control functions. Where this is done, all equipment common to both the protection and control circuits are classified as part of the protection system. Isolation amplifiers prevent a control system failure from affecting the protection system. In addition, where failure of a protection system component can cause a process excursion which requires protective action the protection system can withstand another independent failure without loss of function. Generally, this is accomplished with two-out-of-four trip logic. Also, wherever practical,

provisions are included in the protection system to prevent a plant outrage because of single failure of a sensor.

Specific Control and Protection Interactions

Nuclear Flux

Four power range nuclear flux channels are provided for nuclear overpower protection. Isolated outputs from all four channels are averaged for automatic control rod regulation of power. If any channel fails in such a way as to produce a low output, that channel is incapable of proper nuclear overpower protection. In principle, the same failure would cause rod withdrawal and overpower. Two-out-of-four nuclear overpower trip logic will ensure a nuclear overpower trip if needed even with an independent failure in another channel.

In addition, the control system will respond only to rapid changes in indicated nuclear flux; slow changes or drifts are overridden by the temperature control signals. Also, a rapid decrease of any nuclear flux signal will block automatic rod withdrawal as part of the rod drop protection circuitry.

Finally, an overpower signal from any nuclear channel will block automatic rod withdrawal. The set point for this rod stop is below the reactor trip set point.

Coolant Temperature

Four T_{avg} channels are used for overtemperature-overpower protection. (See Plant Drawings IP3V-0171-0052, -0053, -0054, and -0055 [Formerly Figure 7.2-12] for single channel). Isolated output signals from all four channels are also averaged for automatic control rod regulation. In principle, a spuriously low temperature signal from one sensor could cause rod withdrawal and overtemperature. Two-out-of-four overtemperature and overpower ΔT logic will ensure a trip is needed even with an independent failure in another channel. In addition, channel deviation alarms in the control system will block automatic rod withdrawal if any temperature channel deviates significantly from the others. Automatic rod withdrawal blocks will also occur if any one of four nuclear channels indicates an overpower condition or if any one of the four temperature channels indicates an overtemperature condition. Finally, as shown in Section 14.1, the combination of trips on nuclear overpower, high pressurizer water level, and high pressurizer pressure also serve to limit an excursion for any rate of reactivity insertion.

Narrow range RCS hot leg temperature is measured for each channel through the use of three RTDs located 120° apart. The three RTD signals are averaged by a microsprocessor to produce the hot leg signal for the channel. The microprocessor has the capability to detect a failure of any of the hot leg RTDs.

Pressurizer Pressure

Four pressure channels are used for high and low pressure protection and for overpower-overtemperature protection. Three of these are also used for high pressure protection. Isolated output signals from these channels are also used for pressure control. These are discussed separately below:

1) Pressure Control. Spray, power-operated relief valves, and heaters are controlled by isolated output signals from the pressure protection channels:

   a) Low Pressure

   A spurious high pressure signal from one channel can cause low pressure by actuation of a pressurizer spray valve. Spray reduces pressure at a low rate, and some time is available for operator action (about three minutes at maximum spray rate) before a low pressure trip is reached. Additional redundancy is provided by the protection system to ensure underpressure protection, i.e., two-out-of-four low pressure reactor trip logic and two-out-of-three safety injection logic.

   Each pressurizer relief valve is interlocked to prevent opening on a single high pressure signal. Furthermore, the valve setpoint is at a higher pressure than the normal high pressure signal actuation pressure.

   b) High Pressure

   The pressurizer heaters are incapable of overpressurizing the Reactor Coolant System. Maximum steam generation rate with heaters is about 15,000 lbs/hr, compared with a total capacity of 1,260,000 lbs/hr for the three safety valves and total capacity of 358,000 lbs/hr of the two power-operated relief valves. Therefore, overpressure protection is not required for a pressure control failure. Two-out-of-three high pressure trip logic is therefore used.

   In addition, either of the two relief valves can easily maintain pressure below the high pressure trip point. The two relief valves are controlled by independent pressure channels, one of which is independent of the pressure channel used for heater control. Finally, the rate of pressure rise achievable with heaters is slow, and ample time and pressure alarms are available for operator action.

   An Overpressure Protection System prevents the reactor vessel pressure from exceeding the Technical Specification limits, as described in Section 4.3.4.

   c) Pressurizer Level

   The presssurizer level channels are used for high level reactor trip two out of three. Isolated output signals from these channels are used for volume control, increasing or decreasing water level. A level control failure could fill or empty the pressurizer at a slow rate (on the order of half an hour or more).

2) High Level

A reactor trip on pressurizer high water level is provided to prevent rapid thermal expansions of reactor coolant fluid from filling the pressurizer; the rapid change from high rates of steam relief to water relief can be damaging to the safety valves and relief piping and pressure relief tank. However, a level control failure cannot actuate the safety valves because the high pressure reactor trip is set below the safety valve set pressures. Therefore, a control failure does not require protection system action. In addition, ample time and alarms are available for operator action.

3) Low Level

For control failures which tend to empty the pressurizer, a low level signal from either of two independent level control channels will isolate letdown, thus preventing the loss of coolant. Ample time and alarms exist for operator action.

A low pressurizer level will result for all Loss-of-Coolant Accidents except for a special class of breaks in the range of 2 to 6 inches which occur in the vapor space of the pressurizer. For this special class which does not result in low pressurizer water level, the reactor will be tripped on either low pressure or DT overtemperature as the pressure drops, and DNB will be prevented. Following reactor trip, there will be no core damage as long as the core remains covered. Sufficient time is available in accidents of this type for the operator to take manual control of makeup to assure core cooling during subsequent cold shutdown procedures.

Sufficient redundancy is provided to accommodate the loss of one level channel without jeopardizing functional capability of the reactor protection system. In the Technical Specifications, limits are set on the minimum number of operable channels and required plant status for all reactor protection instrumentation.

Steam Generator Water Level; Feedwater Flow

Before describing control and protection interaction for these channels, it is beneficial to review the protection system basis for this instrumentation.

The basic function of the reactor protection circuits associated with low steam generator water level and low feed water flow is to preserve the steam generator heat sink for removal of long term residual heat. Should a complete loss of feedwater occur with no protective action, the steam generators would boil dry and cause an overtemperature-overpressure excursion in the reactor coolant. Reactor trips on temperature, pressure, and pressurizer water level will trip the plant before there is any damage to the core or reactor coolant system. However, residual heat generated after the reactor trip would cause a pressure spike in the pressurizer that lifts the pressurizer relief valves and causes discharge of liquid reactor coolant to the Containment.
Redundant auxiliary feedwater pumps are provided to prevent this. Reactor trips act before the steam generators are dry to reduce the required capacity and starting time requirements of these pumps and to minimize the thermal transient on the reactor coolant system and steam generators. Independent trip circuits are provided for each steam generator for the following reasons:

1) Should severe mechanical damage occur to the feedwater line to one steam generator, it is difficult to ensure the functional integrity of level and flow instrumentation for that unit. For instance, a major pipe break between the feedwater flow element and the steam generator

would cause high flow through the flow element. The rapid depressurization of the steam generator would drastically affect the relation between downcomer water level and steam generator water inventory.

2) It is desirable to minimize thermal transient on a steam generator for credible loss of feed water accidents.

It should be noted that controller malfunctions caused by a protection system failure affect only one steam generator. Also, they do not impair the capability of the main feedwater system under either manual control or automatic Tavg control. Hence, these failures are far from being the worst case with respect to decay heat removal with the steam generators.

a) Feedwater Flow

A spurious high signal from the feedwater flow channel being used for control would cause a reduction in feedwater flow and prevent that channel from tripping. A reactor trip on low-low water level, independent of indicated feedwater flow, will ensure a reactor trip if needed.

In addition, the three-element feedwater controller incorporates reset on level, such that with expected gains, a rapid increase in the flow signal would cause only a 12-inch decrease in level before the controller reopened the feedwater valve. A slow increase in the feedwater signal would have no effect at all.

b) Steam Flow

A spurious low steam flow signal would have the same effect as a high feedwater signal, discussed above.

c) Level

A spurious high water level signal from the protection channel used for control will tend to close the feedwater valve. This level channel is independent of the level and flow channels used for reactor trip on low flow coincident with low level.

1) A rapid increase in the level signal will completely stop feedwater flow and actuate a reactor trip on low feedwater flow coincident with low level.

2) A slow drift in the level signal may not actuate a low feedwater signal. Since the level decrease is slow, the operator has time to respond to low level alarms. Since only one steam generator is affected, automatic protection is not mandatory and reactor trip on two out of three low-low level is acceptable.

7.2.4 Qualification Testing

Typical protection system equipment is subjected to type tests under simulated seismic acceleration to demonstrate its ability to perform its functions. Type testing is performed using conservatively large accelerations and applicable frequencies. The peak accelerations and frequencies used are checked against those derived by structural analysis of operational and design basis earthquake loadings. Typical switches and indicators for safety features

components have been tested to determine their ability to withstand seismic forces without malfunction which would defeat automatic operation of the required component.

For testing there is no adequate way of knowing what combination of vertical and horizontal input motion produces the worst effects (e.g., stresses, deflections). There is a greater probability that due to the phase relationship of the two simultaneously applied input motions, the resulting combined motion produces less severe effects than when these motions are applied separately. Testing in one direction at a time is considered the best way to obtain positive proof of the equipment's capability. (The independent testing in each of the three directions is also recommended in the IEEE Guide for Seismic Qualifications of Class I Electric Equipment.) Furthermore, the uni-directional testing was performed in a conservative manner, thus providing a margin against any greater effects which may possibly result from the worst combination of simultaneous testing. These conservatisms consist of: (1) an input sine beat motion with 10 cycles per beat, (2) resonant testing at all determined and applicable natural frequencies, (3) further testing at other selected frequencies, and (4) high input acceleration values, particularly for the vertical direction.

Qualification testing was performed on various safety systems such as process instrumentation and nuclear instrumentation. This testing involved demonstrating operation of safety functions at elevated ambient temperatures to 120°F for original control room equipment.

To establish the combined effect upon protection systems of long term operation followed by exposure to accident conditions inside the containment, selected components were subjected to thermal aging followed by irradiation. In addition, components were first irradiated and then subjected to thermal aging. Results of the tests indicate that the components would perform satisfactorily following a Design Basis Accident.

Cables of the type used for Indian Point 3 were tested using the same approach as described above, i.e., irradiation, thermal aging followed by steam exposure and thermal age, and irradiation followed by steam exposure. During exposure to steam, the cables carry nominal voltage and current.

Westinghouse Topical Reports, WCAP-7817(1), WCAP-7817 Supplement 1(2), and WCAP-8234(3) provide the seismic evaluation of safety related equipment. The type tests covered by these reports are applicable to Indian Point 3.

References

1) Vogeding, E. L., "Seismic Testing of Electrical and Control Equipment," WCAP-7817, December 1971.

2) Vogeding, E. L., "Seismic Testing of Electrical and Control Equipment (WCID Process Control Equipment)," WCAP-7817 Supplement 1, December 1971.

3) "Seismic Testing and Functional Verification of By-Pass Loop Reactor Coolant RTD's," WCAP-8234 (Westinghouse Non-Proprietary Class 3), June 1974.

4) NSE 94-3-124 ED, Rev. O "Evaluation of Cable Channelization Deficiencies."

Table 7.2-1

LIST OF REACTOR TRIPS & CAUSES OF ACTUATION OF:
ENGINEERED SAFETY FEATURES, CONTAINMENT AND STEAM LINE ISOLATION & AUXILIARY FEEDWATER

	COINCIDENCE CIRCUITRY AND INTERLOCKS	COMMENTS
REACTOR TRIP		
1) Manual	1/2, no interlocks	
2) Overpower nuclear flux	2/4	High and low settings; manual block and automatic reset of low setting by P-10, Table 7.2-2
3) Overtemperature !T	2/4, no interlocks	
4) Overpower !T	2/4, no interlocks	
5) Low pressurizer pressure	2/4, blocked by P-7	
6) High pressurizer pressure (fixed set points)	2/3, no interlocks	
7) High pressurizer water level	2/3, blocked by P-7	
8) a. Low reactor coolant flow	2/3, per loop, blocked by P-7, P-8	
b. Reactor coolant pump breaker	1/1, per loop, blocked by P-7, P-8	Reactor coolant pump breaker is tripped on underfrequency
c. Undervoltage on reactor coolant pump bus	2/4, per loop, blocked by P-7	
d. Underfrequency on reactor coolant pump bus	2/4	Underfrequency trips all reactor coolant pumps
9) Safety injection signal (Actuation)	2/3, low pressurizer pressure (manual block permitted by 2/3 low pressurizer pressure): or 2/3 high containment pressure (high-level): or 2/3 high differential pressure between any two steam lines, or manual 1/2, or two sets of 2/3 hi-hi containment pressure (high-high pressure) [energize to actuate], or after delay (maximum 6 seconds) with high steam flow in 2/4 lines coincidence with (a) low T_{avg} in 2/4 lines or (b) low steam line pressure in 2/4 lines	
10) Turbine generator	2/3, blocked by P-8	Low auto-stop oil pressures signal
11) Steam/feedwater flow mismatch	1/2 steam/feedwater flow mismatch in selected coincidence with low steam generator water level	

Table 7.2-1

LIST OF REACTOR TRIPS & CAUSES OF ACTUATION OF:
ENGINEERED SAFETY FEATURES, CONTAINMENT AND STEAM LINE ISOLATION & AUXILIARY FEEDWATER

	COINCIDENCE CIRCUITRY AND INTERLOCKS	COMMENTS
	in that steam generator	
12) Low-low steam generator water level	2/3, per loop	
13) High intermediate range nuclear flux	1/2 manual block permitted by P-10	Manual block and automatic reset
14) High source range nuclear flux	1/2 manual block permitted by P-6, block maintained by P10	Manual block and automatic reset of P-6; manual reset of P-10
CONTAINMENT ISOLATION ACTUATION		
15) Safety Injection Signal (Phase A)	See Item 9	Actuates all non-essential service containment isolation trip valve and actuates Isolation Valve Seal Water System
16) Containment pressure (Phase B)	Coincidence of two sets of 2/3 containment pressure (High-high pressure [energize to actuate], same signal which actuates containment spray), or manual 2/2	Actuates all essential service containment isolation trip valves
17) Containment ventilation (High containment activity)	1/2 high activity signal, from air particulate detector or radiogas detector or containment isolation phase “A” signal, or spray actuation signal	This additional signal closes containment purge supply, exhaust ducts and pressure relief duct only
ENGINEERED SAFETY FEATURES ACTUATION		
18) Safety injection signal (S)	See Item 9	
19) Containment spray signal (P)	Coincidence of two sets of 2/3 containment pressure (high-high pressure); or manual 2/2 (Note: Bistables are energize-to-operate)	
20) Deleted		
21) Containment air recirculation cooling and filtration signal	Safety injection signal initiates starting of all fans in accordance with the safety injection starting sequence, 2/3 high containment pressure or manual 1/2	
22) Isolation valve seal water signal	Safety injection signal	
STEAM ISOLATION ACTUATION		
23) Steam flow	After time delay (maximum 6 seconds) with high	

Table 7.2-1

LIST OF REACTOR TRIPS & CAUSES OF ACTUATION OF:
ENGINEERED SAFETY FEATURES, CONTAINMENT AND STEAM LINE ISOLATION & AUXILIARY FEEDWATER

	COINCIDENCE CIRCUITRY AND INTERLOCKS	COMMENTS
	steam flow in 2/4 lines in coincidence with (a) low T_{avg} in 2/4 lines or (b) low steam line pressure in 2/4 lines	
24) Containment pressure	Coincidence of two sets of 2/3 Containment pressure (high-high pressure) (Note: Bistables are energize-to-operate)	
25) Manual	1/1 per steam line	
AUXILIARY FEED WATER ACTUATION		
26) Turbine driven pump	Coincidence of 2/3 low level in two steam generators; or a non-SI blackout sequence signal from 480 volt buses 3A or 6A; or manual 1/2 or AMSAC Actuation	
27) Motor driven pumps	2/3 low level in any steam generator; or trip of ½ main feedwater pump turbines; or safety injection signal; or manual 1/2; or a non-SI blackout sequence signal from 480 volt bus 3A to start pump 31; or a non-SI blackout sequence signal from 480 volt bus 6A to start pump 33; or AMSAC Actuation	
MAIN FEEDWATER ISOLATION		
28) Close main feedwater control valves, (including associated MOVs) trip main feedwater pumps	Any safety injection signal (See Item 9)	

TABLE 7.2-2

INTERLOCK AND PERMISSIVE CIRCUITS

Number	Function	Input for Blocking
1 +	Prevent rod withdrawal on overpower	1/4 high nuclear flux (power range) or 1/2 high nuclear flux (intermediate range or 1/4 overtemperature ΔT or 1/4 overpower ΔT
2	Auto-rod withdrawal stop at low power	Low MWe load signal
3+	Auto-rod withdrawal stop on rod drop	1/4 rapid decrease of nuclear flux (power range) or 1/1 rod bottom indication
4*	[BLANK – See Note]	
5 +	Steam dump interlock	Turbine trip signal
6	Manual block of source range level trip	1/2 high intermediate range flux allows manual block, 2/2 low intermediate range defeats block
7	Permissive power (block various trips required only at power)	3/4 low-low nuclear flux (power range) and 2/2 low turbine impulse chamber pressure signal
8	Block single primary loop loss of flow trip and Block Reactor Trip on Turbine Trip	3/4 low nuclear flux (power range)
9*		
10	Manual block of low setpoint trip (power range) and intermediate range trips	2/4 high nuclear flux allows manual block, 3/4 low nuclear flux (power range) defeats manual block

NOTE:
* not applicable to this plant
+ alarmed

TABLE 7.2-3

ROD STOPS

Rod Stop	Actuation Signal	Rod Motion to be blocked
1. Rod Drop	1/4 rapid power range nuclear flux decrease or any rod bottom signal	Automatic Withdrawal [Deleted]
2. Nuclear Overpower	1/4 high power range nuclear flux or 1/2 high intermediate range nuclear flux	Automatic and Manual Withdrawal
3. High ΔT [Deleted]	1/4 overpower ΔT or 1/4 overtemperature ΔT	Automatic and Manual Withdrawal
4. Low Power	Low turbine first stage (inlet) pressure load signals	Automatic Withdrawal
5. T_{avg} Deviation	1/4 T_{avg} deviation from average T_{avg}	Automatic Withdrawal

[Deleted]

TABLE 7.2-4

TABLE OF MAIN CONTROL BOARD INDICATOR AND/OR RECORDERS "*ORIGINALLY*" AVAILABLE TO THE OPERATOR [HISTORICAL]

PARAMETER	NO. OF CHANNELS AVAIL	NO. OF CHANNELS REQUIRED	RANGE	ACCURACY REQUIRED	INDICATOR/RECORDER	PURPOSE
MODERATE & INFREQUENT FAULTS						
1. TCold or Thot (Measured, Wide range)	4THot 4 TCold	1	0-700°F	+4% of Full range	Both channels are recorded on each loop.	Ensure maintenance of proper cooldown rate to ensure maintenance of proper relationship between system pressure and temperature for NDTT considerations.
2. Pressurizer Water Level	3	2	Entire Distance Between Taps	+3% of Level at 2250 PSIA	All 3 channels indicated; one channel is selected for recording.	Ensure maintenance of proper reactor coolant inventory.
3. Reactor Coolant System Pressure (Wide range)	2	1	0-3000 psig	+6% of Full range	Indicated and recorded	Ensure maintenance of proper relationship between system pressure and temperature for NDTT consideration.

4. Containment Pressure	6	1	-5 psig to +75 psig	±3% of Full range	All 6 are indicated.	Monitor containment conditions to indicate need for potential safeguards actuation.
5. Steam Line Pressure	12 (3/Loop)	4 (1/Loop)	0-1400 psig	±3% of Full Scale	All 4 are indicated.	Monitor steam generator temperature conditions during hot shutdown and cooldown and for use in recovery from steam generator tube ruptures.
6. Steam Generator Water Level (Wide range)	4 (1/S.G.)	*	+7 to -41 feet from nominal full load water level	±5% of Level Span (Cold)	All 4 channels recorded.	Ensure maintenance of reactor heat sink.
7. Steam Generator Water Level (Narrow range)	12 (3/S.G.)	*	+7 to -5 feet from nominal full load water level	±3% of Level Span (Hot)	All 12 Channels indicated; the 4 channels used for control are recorded.	Same as 6.

Minimum Requirements: One level channel per Steam Generator (Either Wide or Narrow Range) with at least Two Wide Range Channels

TABLE 7.2-5

TABLE OF MAIN CONTROL BOARD INDICATOR AND/OR RECORDERS *"ORIGINALLY"* AVAILABLE TO THE OPERATOR [Historical]

Parameter	No. of Channels Avail	Required	Range	Accuracy Required	Indicator/Recorder	Purpose
Limiting Faults						
1. Containment Pressure	6	1	-5 psig to +75 psig	±10% of Full Scale	All 6 are indicated	Monitor Post-LOCA containment conditions.
2. Refueling Water Storage Tank Water Level	2	1	0-100% of span	±3% of level span	One is indicated and both are alarmed	Ensure that water is flowing to the safety injection system after a LOCA and determine when to shift from injection to recirculation mode.
3. Steam Generator Water Level (narrow range)	3/Steam Generator	*	+7 to -5 feet from nominal full load level	±10% of level span (1)	All channels indicated; the channels used for control are recorded	Detect steam generator tube rupture; monitor steam generator steam water level following a line break.
4. Steam Generator Water Level (wide range)	1/Steam Generator	*	+7 to -41 feet from nominal full load level	±10% of level span (1)	All channels are recorded	Detect steam generator tube rupture; monitor steam generator water level following a steam line break.

5. Steam Line Pressure	3/Steam line	1/Steam line	0-1400 psig	±5% of full scale	All channels are indicated	Monitor steam line pressures following steam generator tube rupture or steam line break.
6. Pressurizer Water Flow Level	3	2	Entire distance between taps	Indicate that level is somewhere between 0 and 100% of span	All 3 are indicated and one is for recording	Indicate that water has returned to the pressurizer following cooldown after steam generator tube rupture or steam line break.
7. RHR Recirculation Flow	4	3**	0-1000 GPM	±10% of span	All are indicated	Monitor recirculation flow.

(1) For the steam break, when the water level channel is exposed to a hostile environment, the accuracy required can be relaxed. The indication need only convey to the operator that water level in the steam generator is somewhere between the narrow range steam generator water level taps.

* Minimum Requirements: One Level Channel per Steam Generator (either Wide or Narrow Range) with at least Two Wide Range Channels.

** Three required to allow possibility of low head recirculation. None required to allow high head recirculation.

7.3 REGULATING SYSTEMS

7.3.1 Design Basis

The Reactor Control System is designed to limit nuclear plant transients for prescribed design load perturbations, under automatic control, within prescribed limits to preclude the possibility of a reactor trip in the course of these transients.

Overall reactivity control is achieved by the combination of chemical shim and 53 control rod clusters of which 29 are in 4 control banks and 24 are in 4 shutdown banks. Long-term regulation of core reactivity is accomplished by adjusting the concentration of boric acid in the reactor coolant. Short-term reactivity control for power changes or reactor trip is accomplished by movement of control rod clusters.

The primary function of the Reactor Control System is to provide automatic control of the rod clusters during power operation of the reactor. The system uses input signals including neutron flux; coolant temperature and pressure; and plant turbine load. The Chemical and Volume Control System (Chapter 9) serves as a secondary reactor control system by the addition and removal of varying amounts of boric acid solution.

There is no provision for a direct continuous visual display of primary coolant boron concentration. When the reactor is critical, the best indication of reactivity status in the core is the position of the control group in relation to plant power and average coolant temperature. There is a direct, predictable, and reproducible relationship between control rod position and power and it is this relationship which establishes the lower insertion limit calculated by the rod insertion limit monitor. There are two alarm setpoints to alert the operator to take corrective action in the event a control bank approaches or reaches its lower limit. Rod position is also a function of core life.

Any unexpected change in the position of the control banks when under automatic control or a change in coolant temperature when under manual control provides a direct and immediate indication of a change in the reactivity status of the reactor. In addition, periodic samples of coolant boron concentration are taken. The variation in concentration during core life provides a further check on the reactivity status of the reactor including core depletion.

The Reactor Control System is designed to enable the reactor to follow load changes automatically when the plant output is above 15% of nominal power. Control rod positioning may be performed automatically when plant output is above this value, and manually at any time.

Overriding the rod stop and turbine runback signals from the Overpower or Overtemperature ΔT circuitry, or from the Power Range Nuclear Instrument Dropped Rod circuitry has no impact on the prevention of automatic control rod withdrawal below 15% of nominal power. Overriding one channel of these signals has no impact on reactor protection in the event of an approach to an overpower condition in as much as the reactor trips associated with such a condition remains unaffected. Additionally, since only one channel at a time is permitted to be affected, the other three channels remain available for rod stop and turbine runback on either Overpower or Overtemperature ΔT, or on Power Range Nuclear Instrument Rod Drop signals.

The system enables the nuclear plant to accept the following transients without reactor trip subject to possible xenon limitations:

a) Step load increases to 10% within the load range of 15% to 90% of full power
b) Step load reduction of 10% within the load range of 100% to 25% of full power
c) A 5% per minute ramp load change within the load range of 15% to 100% of full power.

The operator is able to select any single bank of rods (shutdown or control) for manual operation. Using a single switch, he may not select more than one bank from these two functions. He may also select automatic reactor control, in which case, the control banks can be moved only in their normal sequence with some overlap as one bank reaches its full withdrawal position and the next bank begins to withdraw. Interlocks are provided to preclude simultaneous withdrawal of more than two banks of control rods or shutdown rods.

The control system is capable of restoring coolant average temperature to within the programmed temperature deadband, following a scheduled or transient change in load.

The reactor plant can be placed under automatic control in the power range between 15 percent of load and full load and will accept the following design transients while in automatic control:

a) Step load increases of 10% within the load range of 15% to 90% of full power (without turbine bypass)
b) Step load reductions of 10% within the load range of 100% to 25% of full power (without turbine bypass)
c) A 5% per minute ramp load change within the load range – 15% to 100% of full power (without turbine bypass)
d) A -10% to -50% change in load, at a maximum turbine unloading rate of 200% per minute, from approximately 100% load with steam dump (load rejection capability depends on full power Tavg; see Section 7.3.2) (with turbine bypass).

A programmed pressurizer water level as a function of T_{avg} is provided to minimize the requirements of the Chemical and Volume Control and Waste Disposal Systems resulting from coolant density changes during loading and unloading from full power to zero power.

Following a reactor and turbine trip, sensible heat stored in the reactor coolant is removed without actuation of steam generator safety valves by means of controlled steam bypass to the condenser and by injection of feedwater to the steam generators. Reactor Coolant System temperature is reduced to the no load condition. This no load coolant temperature is maintained by steam bypass to the condensers to remove residual heat.

The control system is designed to operate as a stable system over the full range of automatic control throughout core life without requiring operator adjustment of set points other than normal calibration procedures.

7.3.2 System Design

A block diagram of the Reactor Control System is shown in Figure 7.3-1.

Rod Control

There are 53 total RCC assemblies. The assemblies are grouped into (1) 4 shutdown banks having rod clusters of 8, 8, 4, 4, rod clusters and (2), 4 control banks 8, 4, 8 and 9 rod clusters.

Figure 3.2-1 shows the location of the RCC assemblies in the core. The four control banks are the only rods that can be manipulated under automatic control. The banks are divided into groups to obtain smaller incremental reactivity changes. All RCC assemblies in a group are electrically paralleled to step simultaneously. Position indication for each RCC assembly type is the same.

Control Group Rod Control

The Reactor Control System is capable of restoring programmed average temperature following a scheduled or transient change in load. The coolant average temperature is programmed to increase linearly from zero power to the full power conditions.

The control system will also compensate initially for reactivity changes caused by fuel depletion and/or xenon transients. Final compensation for these two effects is periodically made with adjustments of boron concentration. The control system then readjusts the control rods in response to changes in coolant average temperature resulting from changes in boron concentration.

The coolant average temperatures are measured from the hot leg and the cold leg in each reactor coolant loop. The average of the four measured average temperatures is the main control signal. This signal is sent to the control rod programmer through a proportional plus rate compensation unit. The control rod programmer commands the direction and speed of control rod motion. A compensated pressurizer pressure signal, and a power-load mismatch signal are also employed as control signals to improve the plant performance. The power-load mismatch channel takes the difference between nuclear power (average of all four power range channels) and a signal of turbine load (first stage (inlet) turbine pressure), and passes it through a high-pass filter such that only a rapid change in flux or power causes rod motion. The pressure compensation and the power-load mismatch compensation serve to speed up system response and to reduce transient peaks.

The control bank rods are divided into four banks comprising 8, 4, 8 and 9 RCC assemblies respectively, to follow load changes over the full range of power operation. Each control rod bank is driven by a sequencing, variable speed rod drive control unit. The assemblies in each control bank are divided into two groups. The groups are moved sequentially one step at a time. The sequence of motion is reversible, that is, a withdrawal sequence is the reverse of the insertion sequence. The variable speed sequential rod control affords the ability to insert a small amount of reactivity at low speed to accomplish fine control of reactor coolant average temperature about a small temperature deadband. Any reactor trip signal causes the rods to drop by gravity into the core.

Manual control is provided to manually move a control bank in or out at a preselected fixed speed.

Proper sequencing of the RCC assemblies is assured: first, by fixed programming equipment in the Rod Control System, and second, through administrative control of the reactor plant operator. Startup of the plant is accomplished by first manually withdrawing the shutdown rod banks to the full out position. This action requires that the operator select the SHUTDOWN BANK position on a control board mounted selector switch and then position the IN-HOLD-OUT level (which is spring return to the HOLD position) to the OUT position.

RCC assemblies are then withdrawn under manual control of the operator by first selecting the MANUAL position on the control board mounted selector switch and then positioning the IN-HOLD-OUT LEVER to the OUT position. In the MANUAL selector switch position, the rods are withdrawn (or inserted) in a predetermined programmed sequence by the automatic programming equipment.

When the reactor power reaches approximately 15% of rated power, the operator may select the AUTOMATIC position, where the IN-HOLD-OUT lever is taken out of service, and rod motion is controlled by the Reactor Control and Protection Systems. A permissive interlock limits automatic control to reactor power levels above 15%. In the AUTOMATIC position, the rods are again withdrawn (or inserted) in a programmed sequence by the automatic programming equipment.

Programming is set so that as the first bank out (control bank A) reaches a preset position, the second bank out (control bank B) begins to move out simultaneously with first bank. When control bank A reaches the top of the core, it stops, and control bank B continues until it reaches a preset position near the top of the core where control bank C motion beings, etc. The withdrawal sequence continues until the plant reaches the desired power level. The programmed insertion sequence is the opposite of the withdrawal sequence, i.e., the last control bank out is the first control bank in.

With the simplicity of the rod program, the minimal amount of operator selection, and two separate direct position indications available to the operator, there is very little possibility that rearrangement of the control rod sequencing could be made.

Shutdown Rod Control

The shutdown rods together with the control rods are capable of shutting the reactor down. They are used in conjunction with the adjustment of chemical shim to provide shutdown margin of at least one percent following reactor trip with the most reactive control rod in the fully withdrawn position for all normal operating conditions. The shutdown banks are manually controlled during normal operation and are moved at a constant speed with staggered stepping of the groups within the banks. Any reactor trip signal causes them to drop by gravity into the core. They are fully withdrawn during power operation and are withdrawn first during startup. Criticality is always approached with the control rods after withdrawal of the shutdown banks. Four shutdown banks with a total of 24 clusters are provided.

Interlocks

The rod control group is used for automatic control and is interlocked with measurements of turbine-generator load to prevent automatic control rod withdrawal below 15% of nominal power. The manual and automatic controls are further interlocked with measurements of neutron flux, ÎT and rod drop indication to prevent approach to an overpower condition.

Rod Drive Performance

The control banks are driven by a sequencing, variable speed rod drive programmer. In each control bank of RCC assemblies, two groups (each containing a small number of RCC assemblies) are moved sequentially in a cycle such that both groups are maintained within one step of each other.

The sequence of motion is reversible, that is, withdrawal sequence is the reverse of the insertion sequence. The sequencing speed is proportional to the control signal from the Reactor Control System. This provides control group speed control proportional to the demand signal from the control system.

The output of two paralleled motor generator (M-G) sets provides power to the rod drive mechanism coils through a solid state control system. Two reactor trip breakers are placed in series with the output of the M-G sets. To permit on-line testing, a bypass breaker is provided across each of the two breakers.

RCCA Position Indication

Two separate systems are provided to sense and display control rod position as described below:

a) Analog System – An analog signal is produced for each individual rod by a linear position transmitter.

   An electrical coil stack is located above the stepping mechanisms of the control rod magnetic jacks, external to the pressure housing, but concentric with the rod travel. When the associated control rod is at the bottom of the core, the magnetic coupling between the primary and secondary coil winding of the detector is small and there is a small voltage induced in the secondary. As the control rod is raised by the magnetic jacks, the relatively high permeability of the lift rod causes an increase in magnetic coupling. Thus, an analog signal proportional to rod position is obtained.

   Direct, continuous readout of every control rod is presented to the operator on individual indicators.

   A deviation monitor alarm is actuated if an individual rod position deviates from its relative bank position by a preselected distance.

   Lights are provided for rod bottom positions for each rod. The lights are operated by bistable devices in the analog system.

b) Digital System – The digital system counts pulses generated in the rod drive control system. One counter is associated with each group of control and shutdown rods. Readouts of the digital system are in the form of electromechanical add-subtract counters reading the number of steps of rod movement with one display for each group. These readouts are mounted on the control panel.

The digital and analog systems are separate systems; each serves as backup for the other. Operating procedures require the reactor operator to compare the digital and analog readings upon recognition of any apparent malfunction. Therefore, a single failure in rod position indication does not in itself lead the operator to take erroneous action in the operation of the reactor.

Full Length Rod Drive Power Supply

The full length control rod drive power supply concept, using a single trip bus system, has been successfully employed on all Westinghouse PWR Plants. Potential fault conditions with a single trip bus system are discussed in this section. The unique characteristics of the latch type

mechanisms with its relatively large power requirements makes this system with the redundant series trip breakers particularly desirable.

The solid state rod control system is operated from two parallel connected 438 kVA generators which provide 260 volt line to line, three phase, four wire power to the rod control circuits through two series connected reactor trip breakers.

This AC power is distributed from the trip breakers to a line-up of identical solid state power cabinets and a DC holding cabinet using a single overhead run of enclosed bus duct which is bolted to and therefore comprises part of the power cabinet arrangement. Alternating current from the motor-generator sets is converted to a pulsed direct current by the power cabinet and is then distributed to the mechanism coils. Each complete rod control system includes a single 125/70 volt DC power supply which is used for holding the mechanisms in position during maintenance of normal power supply.

This 125/70 volt supply, which receives its input from the AC power source downstream of the reactor trip breakers, is distributed to each power cabinet and permits holding mechanisms in groups of four by manually positioning switches located in the power cabinets. The 70/40 ampere output capacity limits the holding capability to eight rods.

Reactor Trip

Current to the mechanisms is interrupted by opening either of the reactor trip breakers. The 125/70 volt DC maintenance supply will also be interrupted since this supply receives its input power through the reactor trip breakers.

Trip Breaker Arrangement

The trip breakers are arranged in the reactor trip switchgear in individual metal enclosed compartments. The 1000 ampere bus work, making up the connections between trip breakers are separated by metal barriers to prevent the possibility that any conducting objects could short circuit, or bypass, trip breaker contacts.

Maintenance Holding Supply

The 125/70 volts DC holding supply and associated switches have been provided to avoid the need for bringing a separate DC power source to the rod control system during maintenance on the power cabinet circuits. This source is adequate for holding a maximum of five mechanisms and satisfies all maintenance holding requirements.

Control System Construction

The rod control system equipment is assembled in enclosed steel cabinets. Three phase power is distributed to the equipment through a steel enclosed bus duct, bolted to the cabinets. DC power connections to the individual mechanisms are routed to the reactor head from the solid state cabinets through insulated cables, enclosed junction boxes, enclosed reactor containment penetrations, and sealed connectors. In view of this type of construction, an accidental connection of either an AC or DC power source, either internal or external to the cabinets, is not considered credible.

AC Power Connections

The three phase four wire supply voltage required to energize the equipment is 260 volts line to line, 58.2 Hz, 438 kVA capacity, zig-zag connected. It is unlikely that any power supply, and in particular one as unusual as this four wire power source could be accidentally connected, in phase, in the required configuration. Also it should be noted that this requires multiple connections, not single connections. The closest outside sources available in the plants are 480 volt auxiliary power source and 208 volt lighting source.

Connections of either a 480 or 208 volt, 60 Hz source to the single AC bus supplying the rod control system causes currents to flow between the sources due to an out of phase condition. These currents flow until the generator accelerates to a speed synchronous with the 60 Hz outside source, a time sufficient to trip the generator breakers. The out-of-phase currents for an unlimited capacity outside source, an outside source with a capacity equivalent to the normal generator kVA, and for either one or two M-G sets in service are tabulated below:

Out of Phase Currents (Amperes)

	One M-G Set in Service	Two M-G Sets In Service
480 volts		
Unlimited Capacity	25,000	50,000
438 kVA Capacity	12,000	25,000
208 volts		
Unlimited Capacity	16,000	32,000
438 kVA Capacity	8,000	16,000

All of the foregoing currents are sufficiently high to trip out the generator breakers on either overcurrent or reverse current. This trip-out is detectable by annunciation in the Control Room. If the outside power source trips, the connection is of no concern.

Each solid state power cabinet is tied to the main AC bus through three fused disconnect switches; one for the stationary gripper coil circuits, one for the movable gripper coil circuits, and one for the lift coil circuits. Reference voltage to operate the control circuits for all three coil circuits must be in phase with the supply to all coil circuits for proper operation of the system. If the outside power source were brought into an individual cabinet, nine (9) normal source connections would have to be disconnected and the outside source would have to be tied in phase to the proper nine (9) points plus one (1) neutral point to allow movement of the rods. This is not considered credible.

Connection of a single phase AC source (i.e., one line to neutral) is also considered improbable. This would again require a high capacity source which would have to be connected in-phase with the non-synchronous M-G set supply. Again, more than one connection is needed to achieve this condition. Each power cabinet contains three alarm circuits (stationary, movable and lift) that would annunciate the condition to the operator. In addition, calculations show that a single phase source of 208 volts, 260 volts, or 480 volts will not supply enough current to hold

the rods. Therefore, a jumper across two trip circuit breaker contacts in series which results in a single phase remaining closed would not provide sufficient current to hold up the rods.

The normal source generators are connected in a zig-zag winding configuration to eliminate the effects of direct current saturation of the machines resulting from the direct currents that flow in the half wave bridge rectifier circuits. If this connection were not used, the generator core would saturate and loss of generating action would occur. This condition would also occur in a transformer. An outside source not having the zig-zag configuration would have to have a large capacity (400 kVA) to avoid the loss of transformer action from saturation.

Most of the components in the equipment are applied with a 100% safety factor. Therefore, the possibility exists that the system will operate at 480 volts with a source of sufficient capacity. The system will definitely operate at 208 volts with a source of sufficient capacity.

The connection of an outside source of AC power to one rod control system would first require a need for this source. No such need exists since two power sources (M-G sets) are already provided to supply the system. If the source were connected in spite of the need, extreme measures would have to be taken by the intruder to complete the connection. The outside source would have to be a large capacity (400 kVA) one. The currents that flow would require the routing of large conductors or bus bars, not the usual clip leads. Then the disassembly of switchgear or enclosed bus duct would be required to expose the single AC bus. Large bolted cable or bus bar terminations would have to be completed. A total of four conductors would have to be connected in phase with a non-synchronous source. To expect that a connection could be completed with the equipment either energized or de-energized in view of the obstacles which would prevent such a connection is incredible.

However, even if the connection were completed, the outside source connection would be detectable by the operator through the tripping of the generator breakers.

DC Power Connections

An external DC source could, if connected inside the power cabinet, hold the rods in position. This would require a minimum supply voltage of 50 volts. Since the holding current for each mechanism coil is 4 amperes, the DC current capacity would have to be approximately 180 amperes to hold all rods. Achieving this situation would require several acts – bringing in a power source which is not required for any type of operation in the rod control system, preferentially connecting it into the system at the correct points, and actuating specific holding switches so as to interconnect all rods. Closure of twelve switches in four separate cabinets would be required to hold all rods. One switch could hold as many as four rods.

The application of a DC voltage to an individual rod external to the power cabinet would affect only a single rod connection with other rods in the group being prevented by the blocking diodes in the power circuits.

Should an external DC source be connected to the system, the system is provided with features to permit its detection.

Each solid state power cabinet contains circuitry which compares the actual currents in the stationary and movable gripper coils with the reference signals from the step sequencing unit (slave cycler). In taking a single step, the current to the stationary gripper coil will be profiled from the holding value to the maximum, to zero and return to holding level. Correspondingly,

the movable gripper coil must change from zero to maximum and return to zero. The presence of an external DC source on either the stationary or movable would prevent the related currents from returning to zero.

This situation would be instantaneously annunciated by way of the comparison circuit. Therefore, any rod motion would actuate an alarm indicating the presence of an external DC source. In addition, an external DC source would prevent rods from stepping. Thus, an external source could be detected by the rod position indication system indicating failure of the rod(s) to move.

Connection of an external DC power source to the output lines of the 125/70 volt DC power supply can be detected by opening the three phase primary input of the supply and checking the output indication lights.

Evaluation Summary

In view of the preceding discussion, the postulated connection of an external power source (either AC or DC) or occurrence of short circuits that could prevent dropping of the rods is not considered credible.

Specifically:

a) The need for an outside power source has been eliminated by incorporating built-in holding sources as part of the rod control system and by providing two M-G sets.

b) The equipment is contained within enclosed steel cabinets precluding the possibility of an accidental connection of either AC or DC power in the cabinets.

c) AC power distribution is accomplished using steel enclosed bus duct. The high capacity (438 kVA) AC power source is unique and not readily available. Multiple connections are required.

d) DC power is distributed to the individual mechanisms through insulated cables and enclosed electrical connections precluding the accidental connection of an outside DC source external to the cabinets. The high capacity DC source required to hold rods is not readily available in the rod control system, would require multiple connections, and would require deliberate positioning of switches within the enclosed cabinets.

e) Provisions are made in the system to permit detection of an external DC source which could preclude a rod release.

The total capacity of the system including the overload capability of each motor generator set is such that single set out of service does not cause limitations in rod motion during normal plant operation. In order to minimize reactor trip as a result of a unit malfunction, the power system is normally operated with both units in service.

Turbine Bypass

A turbine bypass system is provided to accommodate a reactor trip with turbine trip and in conjunction with automatic reactor control can accommodate a load rejection without reactor and turbine trip. The maximum load rejection that can be accommodated without reactor and turbine trip depends on the full load Tavg. A maximum of a 10% load rejection can be accommodated for the minimum acceptable full load Tavg of 550.6^{o}F. As the full load Tavg is increased, larger load rejections can be accommodated. For full load Tavg values of 565 ^{o}F or higher, load rejections of 50% can be accommodated. The turbine bypass system removes steam to reduce the transient imposed upon the reactor coolant system so that the control rods can reduce the reactor power to a new equilibrium value without allowing overtemperature, overpressure conditions in the Reactor Coolant System.

The steam dump is actuated by an electrical load decrease rate greater than a preset value. This signal supplies air to the dump valves, which then allows them to open and close according to the temperature error signal, a compensated (T_{avg} – T_{ref}) signal. The dump valves modulate open proportionally to this temperature error signal with a stroke time of approximately 20 seconds. For large temperature errors the valves will trip open in two banks as required for fast response with a stroke time of about three seconds. Upon reduction of the error signal below the trip-open setpoints, the respective valve groups return to modulating control.

The steam dump decreases proportionally as the control rods act to reduce the coolant average temperature. The artificial load is therefore removed as the coolant average temperature is restored to its programmed value. When steam dump is no longer required, the air supply to the valves may be manually removed.

Since the steam dump valves exhaust into the condenser, all steam dump is blocked when the condenser in unavailable.

The turbine bypass steam system is described in Section 10.2. The bypass flows to the main condenser.

Feedwater Control

Each steam generator is equipped with two three element feedwater control systems (one for the main regulator valve and the second for the low flow regulator valve) which maintain a programmed water level as a function of load on the secondary side of steam generator. The three element feedwater control system continuously compares actual feedwater flow with steam flow compensated by steam pressure with a water level set point to regulate the feedwater valve opening. The individual steam generators are operated in parallel, both on the feedwater and on the steam side.

Continued delivery of feedwater to the steam generator is required as a sink for the heat stored and generated in the coolant following a reactor trip and turbine trip. A reactor trip signal provides an override signal to the feedwater control system. After a trip, all feedwater valves open fully thereby insuring the full supply of feedwater following a reactor trip and turbine trip. Another override signal then closes the feedwater valves when the coolant average temperature falls below a preset temperature value or when the respective steam generator level rises to a preset value. Manual override of the feedwater control systems is also provided.

Pressure Control

The reactor coolant system pressure is maintained at constant value by using heaters in the water region and spray in the steam region of the pressurizer. Electrical immersion heaters are located near the bottom of the pressurizer. A portion of the heater groups are proportional heaters and are used for small pressure variation control and to compensate for heat losses and the smaller continuous spray. Up to three sets of backup heaters may be turned on manually and operated continuously. The remaining (backup) heaters are turned on either when the pressurizer pressure controller signal is below a preset value or when the pressurizer level exceeds the programmed level setpoint by a preset amount.

The spray valves for the pressurizer are located near their respective RCS cold legs, and the spray nozzle is located at the top of the pressurizer. Spray is initiated when the pressure controller signal is above a preset set point. Spray rate increases proportionally with increasing pressure until it reaches the maximum spray capacity.

Steam condensed by spray reduces the pressurizer pressure. A small continuous spray is normally maintained to reduce thermal stress and thermal shock when the spray valves open and help maintain uniform water chemistry and temperature in the pressurizer.

Two power operated relief valves (PORVs), PCV-455C and PCV-456, prevent the RCS pressure from exceeding the Technical Specifications limits of 10 CFR 50 Appendix “G” during low temperature, low pressure and water solid modes of operation. The PORVs are armed below a preset temperature of 330°F, and will open at a programmed pressure which is set to prevent exceeding the Appendix “G” curves. When the RCS temperature is greater than the LTOP arming temperature (i.e., >330^{o}F) but below the minimum temperature at which the pressurizer safety valves lift prior to violation of the 10 CFR 50, Appendix G, limits (i.e., ≤370^{o}F), administrative controls in the Technical Requirements Manual (TRM) (Ref. 1) are used to limit the potential for exceeding 10 CFR 50, Appendix G, limits. (Ref. 2). The two PORVs are supplied with nitrogen. The instrument N_2 system for the PORVs is tapped from the N_2 supply line to the four safeguards accumulators. The accumulators are sized to provide for 200 valve operating cycles. The actual take-off point for this N_2 system is downstream of the pressure regulator valve NNE-863. The PORV accumulators individually hold 6 cu ft of N_2 at a minimum pressure of 550 psig. During low temperature shutdown operations, the Overpressure Protection System requires an N_2 supply of sufficient capacity which, in case of loss of main N_2 supply, can support the number of PORV cycles resulting from an overpressure event of 10 minute duration. This N_2 supply is provided by one Safety Injection Accumulator having its associated N_2 fill valve blocked open.

One PORV is operated on the pressurizer pressure controller signal, the other one is operated on the actual pressure signal. A separate interlock is provided for each so that if a second pressure channel indicates abnormally low, at the time the relief valve operation is called for by the other channel, the valve activation is blocked. The logic for each is thus basically two out of two. However, during normal operation at normal pressure, the interlock is not actuated and only the operating signals are required to actuate the valve. The interlock is set above normal operating pressure to prevent spurious operation.

Three spring-loaded safety valves limit system pressure to 2750 psia following a complete loss of load without direct reactor trip or actuation of turbine bypass.

Reactor coolant flow to the residual heat removal loop is from the hot leg of Loop 2 through two motor operated valves (No. 731 and 730). Valves 731 and 730 are pressure interlocked to prevent opening should reactor coolant pressure go above 450 psig. This arrangement prevents inadvertent pressurization of the residual heat removal loop when the Reactor Coolant System is above 450 psig. These valves will be opened when RCS pressure is lower than 450 psig. Valve position indication lights and position selector switches for both valves are provided in the control room. These valves are closed during power operation to preclude RHRS over-pressurization. To open the valve, the switch is held over to the Open position and if RCS pressure is less than 450 psig, the valve will open. If these valves are open and RCS pressure increases to 550 psig, they will auto-close. A narrow range pressure recorder with an operator controlled alarm point, which actuates warning lights and audible device, has been added to instrument loop for PT-402. This is designed to attract the operators attention to a potential overpressure transient in progress, to allow him to take necessary action to minimize the magnitude of overpressure event while the RCS is operating at low pressure. The system is not required for safe shutdown of the reactor, and the operator may deactivate the recorder and alarms, which removes the potential for distracting alarms when a normal RCS pressure.

To prevent inadvertent isolation of the RHR loop when the Reactor Coolant System is below 200 degrees, depressurized, and vented to an equivalent opening of greater than two (2) square inches AC-MOV-730 and 731 may be de-energized open.

These valves are also interlocked with containment sump valves 885A and B. To open valves 885A and B, the RHR suction valves 730 and 731, respectively, must be closed. This prevents the reactor coolant water from being drained to the contained sump. High Head SI Suction Valves 888A and B are also interlocked with valves 730 and 731, respectively. A valve 884A and B will not open if 730 and 731, respectively, are opened.

SI-MOV-883 is interlocked with AC-MOV-730 and AC-MOV-731 so that the valve can only be opened if both MOV-730 and MOV-731 are fully closed. If valve SI-MOV-883 is open and valve AC-MOV-730 or AC-MOV-731 leave their closed limit seats, valve SI-MOV-883 will auto-close. The interlock prevents inadvertent opening of valve SI-MOV-883 during cool down and subsequent diversion of reactor coolant to the RWST or over pressurization of a lower pressure SI piping system.

Valves AC-MOV-730 and -731 may be de-energized during cold shutdown if the RCS is depressurized and vented through a minimum equivalent opening of two (2) square inches. De-energizing these valves while the RHR pumps are in service prevents inadvertent isolation of the RHR pump suction supply, which could potentially cause pump failure. De-energizing these valves will also cause a loss of all of the interlock protection associated with AC-MOV-730 and -731. When AC-MOV-730 and –731 are de-energized, administrative controls are established to replace the protective functions of these interlocks. These administrative controls prevent unanticipated communication of reactor coolant with the containment sump and the RWST. These controls also prevent overpressurization of the RHR and SI system piping and components.

7.3.3 System Design Evaluation

Plant Stability

The control system is designed to maintain a stable reactor coolant average temperature within acceptable limits. Continuous oscillation at a low frequency and small amplitude is expected. Proper adjustment of the control loop static and dynamic gains (with respect to the process response) can reduce this oscillation almost to zero and will also avoid instability induced by the control system itself. Because stability is more difficult to maintain at low power under automatic control, no provision is made to provide automatic control below 15 percent of full power.

The control system is designed to operate as a stable system over the full range of automatic control throughout core life.

Step Load Changes Without Turbine Bypass

A typical reactor power automatic control requirement is to restore equilibrium conditions without a plant trip, following 10 percent step load demand increases within the range of 15 to 90 percent of full power and 10% step load demand reductions within the range of 100% to 25% of full power. The design was necessarily based on conservative conditions and a greater transient capability is expected for actual operating conditions. A load demand greater than full power is inhibited by the turbine control load limit devices in response to input from the Reactor Protection System. Although turbine bypass is provided for added control after large load decreases, it will not be necessary during the 10% load changes.

The function of the control system is to minimize the reactor coolant average temperature deviation during the transient within an acceptable value and to restore average temperature to the programmed set point within an acceptable time. Excessive pressurizer pressure variations are prevented by using spray and heaters in the pressurizer.
The margin to over-temperature ΔT reactor trip is of primary concern for the step load changes. This margin is influenced by nuclear flux, pressurizer pressure, and reactor coolant average temperature and temperature rise across the core.

Ramp Loading and Unloading

Ramp loading and unloading is provided over the 15 to 100 percent power range under automatic control. The function of the control system is to maintain the coolant average temperature and the secondary steam pressure as functions of turbine-generator load within acceptable deviation from the programmed values. The minimum control rod speed provides a sufficient reactivity rate to compensate the reactivity changes resulting from the moderator temperature coefficient and the power coefficient.

The coolant average temperature is increasing during loading and there is a continuous in-surge to the pressurizer resulting from coolant expansion. The sprays limit the resulting pressure increase. Conversely, as the coolant average temperature is decreasing during unloading, there is a continuous out-surge from the pressurizer resulting from coolant contraction. The heaters limit the resulting system pressure decrease. The pressurizer level is programmed such that the water level has an acceptable margin above the low level heater cutout set point during the loading and unloading transients.

The primary concern for the loading is to limit the overshoot in coolant average temperature to provide sufficient margin to the over-temperature ΔT trip.

The automatic load controls are designed to safely adjust the unit generation to match load requirements within the limits of the unit capability and licensed rating.

Loss of Load With Turbine Bypass

The Reactor Control System is designed to accept a 10% to 50% (depending on full power Tavg; see Section 7.3.2) loss of load accomplished as a turbine runback at a maximum rate of 200% per minute without requiring a reactor trip. The automatic turbine bypass system is able to accommodate this abnormal load rejection by reducing the thermal transient imposed upon the reactor coolant system. The reactor power is reduced at a rate consistent with the capability of the rod control system. The reducing of the reactor power is automatic down to 15 percent of full power. Manual control is used when the power is below this value. The steam bypass is removed as fast as the control rods are capable of inserting negative reactivity.

The pressurizer relief valves might be actuated for the most adverse conditions, e.g., the most negative Doppler coefficient, and the minimum incremental rod worth. The relief capacity of the power operated relief valves is sized large enough limit the system pressure to prevent actuation of high pressure reactor trip for the most adverse conditions.

Turbine-Generator Trip With Reactor Trip

Turbine-generator unit trip is accompanied by reactor trip. With a secondary system design pressure of 1100 psia, the plant is operated with a programmed average temperature as a function of load, with the full load average temperature significantly greater than the saturation temperature corresponding to the steam generator safety valve set point. This, together with the fact that the thermal capacity in the Reactor Coolant System is greater than that of the secondary system, requires a heat sink to remove heat stored in the reactor coolant to prevent actuation of steam generator safety valves for turbine and reactor trip from full power.

This heat sink is provided by the combination of controlled release of steam to the condenser and by makeup of cold feedwater to the steam generators. The turbine bypass system is controlled from the reactor coolant average temperature signal whose reference set point is reset upon trip to the no load value. Turbine bypass actuation must be rapid to prevent steam generator safety valve actuation. With the bypass valves open the coolant average temperature starts to reduce quickly to the no load set point. The automatic control of reactor coolant average temperature acts to proportionally close the valves and thus minimize the total amount of steam bypassed.

Following turbine trip, the steam voids in the steam generators will collapse and the fully opened feedwater valves will provide sufficient feedwater flow to restore water level in the downcomer. The feedwater flow is cut off if the reactor coolant average temperature decreases below a preset temperature value or if the steam generator water level reaches a preset high set point.

Additional feedwater makeup may then be controlled manually to restore and maintain steam generator level while maintaining the reactor coolant at the no load temperature. Long term residual heat removal is maintained by the steam generator pressure controller (manually selected) which controls the steam pressure (and thus, indirectly, the temperature) by adjusting

the amount of turbine bypass to the condensers. The controller operates the same bypass valves to the condensers which are controlled by coolant average temperature during the initial transient following turbine and reactor trip.

The pressurizer pressure and water level fall very fast during the transient resulting from the coolant contraction. If heaters become uncovered following the trip, the Chemical and Volume Control System will provide full charging flow to restore water level in the pressurizer. Heaters are then turned on to heat up pressurizer water and restore pressurizer pressure to normal.

The turbine bypass and feedwater control systems are designed to prevent the coolant average temperature falling below the programmed no load temperature following the trip to ensure adequate reactivity shutdown margin.

References

1) IP3 Technical Requirements Manual

2) 10 CFR 50, Appendix G

7.4 EXCORE NUCLEAR INSTRUMENTATION

7.4.1 Design Bases

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the basis for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

The Authority has completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

Fission Process Monitors and Controls

Criterion: Means shall be provided for monitoring or otherwise measuring and maintaining control over the fission process throughout core life under all conditions that can reasonably be anticipated to cause variations in reactivity of the core. (GDC 13 of 7/11/67)

The excore Nuclear Instrumentation System is provided to monitor reactor power from source range, through intermediate range and power range, up to 120 percent of full power. The system provides indication, control and alarm signals for reactor operation and protection.

Additionally, per Regulatory Guide 1.97 requirements, an Excore Neutron Flux Monitoring System (NFMS) (see Plant Drawing 9321-LL-96553 [Formerly Figure 7.4-4]) consisting of two detectors has been installed to provide reactor power indication from source range through power range. The Regulatory Guide 1.97 excore Neutron Flux Monitoring System provides local indication elsewhere in the plant, in addition to indication only provided to the control room via QSPDS and Plant Computer. These other indication locations are in the upper electrical tunnel and at the charging station in the PAB for use during shutdown from outside the control room.

The operational status of the reactor is monitored from the Control Room. When the reactor is subcritical (i.e., during cold or hot shutdown, refueling and approach to criticality) the relative reactivity status (neutron source multiplication) is continuously monitored and indicated by proportional counter detectors located in instrument wells in the primary shield adjacent to the reactor vessel. Two source range detector channels are provided for supplying information on multiplication while the reactor is subcritical. A reactor trip is actuated from either channel if the neutron flux level becomes excessive. This system is checked prior to operations in which criticality may be approached. This is accomplished by the use of an incore source to provide a meaningful count rate even at the refueling shutdown condition. Any appreciable increase in the neutron source multiplication is slow enough to give ample time to start corrective action (boron dilution stop and/or emergency boron injection) to prevent the core from becoming critical

When the reactor is critical, means for showing the relative reactivity status of the reactor are:

1) Rod Position
2) Source, Intermediate and Power Range Detector Signals

3) Qualified Safety Parameters Display System (QSPDS)
4) Boron Concentration
5) Hot Leg Temperatures

The position of the control banks is directly related to the reactivity status of the reactor when at power, and any unexpected change in the position of the control banks under automatic control or change in the hot leg coolant temperature under either manual or automatic control provides a direct and immediate indication of a change in the reactivity status of the reactor. Periodic samples of the coolant boron concentration are taken. The variation in concentration during core life provides a further check on the reactivity status of the reactor including core depletion.

High nuclear flux protection is provided both in the power and intermediate ranges by reactor trips, actuated from either range, if the neutron flux level exceeds trip set-points. When the reactor is critical, the best indication of the reactivity status in the core (in relation to the power level and average coolant temperature) are the control room display of the rod control group position and the boron concentration in the coolant.

7.4.2 System Design

Nuclear Instrumentation System (NIS)

The three instrumentation ranges of the Nuclear Instrumentation System (NIS) overlap so that continuous readings are available during transition from one range to another. The sensitivities of the neutron detectors are illustrated on Figure 7.4-1. The Nuclear Instrumentation System diagram is shown on Figure 7.4-2.

Detectors

The excore system consists of twelve independent detectors in six instrument wells located around the reactor, as shown in Figure 7.4-3. The six assemblies provide the following instrumentation:

1. Power Range

   This range consists of four independent, long, uncompensated ionization chamber assemblies. Each assembly is made up of two sensitive lengths. One sensitive length covers the upper half of the core, and the other length covers the lower half of the core.

   In effect the arrangement provides a total of eight separate ionization chambers approximately one-half the core height. The eight uncompensated (guard-ring) ionization chambers sense thermal neutrons in the range from 5.0×10^{2} to 1.0×10^{11} neutrons per sq cm per sec.

   Each chamber initially had a nominal sensitivity of 3.1×10^{-13} amperes per neutron per sq cm (see Figure 7.4-1). The four long ionization chamber assemblies are located in vertical instrument wells adjacent to the four “corners” of the core. The

assembly is manually positioned in the assembly holders and is electrically isolated from the holder by means of insulated standoff rings.

Due to redesign of the Nuclear Core (low leakage core design) and resultant decrease in thermal neutrons at the detectors, new Power Range Moderators have been installed on the four (4) Power Range uncompensated ionization chambers. The Power Range Moderators increase the normal sensitivity of the chambers by approximately 700%.

2. Startup Range (Intermediate and Source)

There are two separate startup range assemblies. Each assembly contains one compensated ionization detector (intermediate range) and one proportional counter detector (source range).

The source range neutron detectors are proportional counters with an initial nominal sensitivity of 10 counts per sec per neutron per sq cm per sec (see Figure 7.4-1). The detectors sense thermal neutrons in the range from 10^{-1} to 5. x 10^{5} neutrons counts per second. The range of the source range channel is 10^{0} to 10^{6} counts per second.

The Source Range detectors are positioned in detector assembly containers by means of a linear, high density moderator insulator. The detector and insulator units are packaged in a housing which is inserted into the detector wells. The detector assembly is electrically isolated from the detector well by means of insulated stand-off rings.

The intermediate range neutron detectors are compensated ionization chambers that sense thermal neutrons in the range form 2.5 x 10^{2} to 2.5 x 10^{10} neutrons per sq cm per sec and initially had a nominal sensitivity of 4 x 10^{-14} amperes per neutron per sq cm per second (see Figure 7.4-1). They produce a corresponding direct current of 10^{-11} to 10^{-3} amp. These detectors are located in the same detector assemblies as the proportional counters for the source range channels.

Other than the source range pre-amplifier, which is located in containment, the electronic components for each of the source, intermediate and power range channels for the NIS are contained in a draw-out- panel mounted in racks in the Control Room.

<u>Power Range Channel</u>

There are three sets of power range measurements. Each set utilizes four individual currents as follows:

a) Four currents directly from the lower sections of the long ionization chambers
b) Four currents directly from the upper sections
c) Four total currents of (a) and of (b), equivalent to the average of each section.

For each of the four currents in (a) and (b), the current measurement is indicated directly by a microammeter, and isolated signals are available for control console indication and recording. An analog signal proportional to individual currents is transmitted through buffer amplifiers to the overtemperature ΔT channel and provides automatic reset of the trip point for these protection functions. The total current, equivalent to the average, is then applied through a linear amplifier

to the bistable trip circuits. The amplifiers are equipped with gain and bias controls for adjustment to the actual output corresponding to 100 percent of rated reactor power.

Each of the four amplifiers also provides amplified isolated signals to the main control board for indication and for use in the Reactor Control System. Each set of bistable trip outputs is operated as a two-out-of-four coincidence to initiate a reactor trip. Bistable trip outputs are provided at low and high power set points depending on the operating power. To provide more protection during startup operation the low range power bistable is used. This trip is manually blocked after a permissive condition is obtained by two of four power range channels. The high power trip bistable is always active.

The overpower trip is set so that, with the maximum instrumentation and bistable set point error, the maximum reactor overpower condition will be limited to 118 percent. This limit is accomplished by the use of solid state instrumentation and long ionization chambers, which permit an integration of the flux external to the core over the total length of the core, thereby reducing the influence of axial flux distribution changes due to control rod motion.

The ion chamber current of each detector is measured by sensitive meters with an accuracy of 0.5 percent. A shunt assembly and switch in parallel with each meter allow selection of one of four meter ranges. The available ranges are 0-100, 0-500, 0-1,000 and 0-5,000 microamperes. The shunt assemblies are designed in such a manner that they will not disconnect the detector current to the summing assembly upon meter failure or during switching. An isolation amplifier provides an analog signal proportional to ion chamber current for recording, data logging and delta flux indication. A test calibration unit provides necessary swtiches and signals for checking and calibrating the power range channels.

The linear amplifier accepts the output currents from each of the two chamber sections and derives a nuclear power signal proportional to the summed direct currents. This unit amplifies the currents and converts the normal current signal to a voltage signal suitable for operation of associated components such as bistables and isolation amplifiers.

Multiple power supplies furnish necessary positive and negative voltages for the individual channels and detector power.

Mounted on the front panel of each power range channel drawer are the ion chamber current meters, the shunt selector switches with appropriate positions, and the nuclear power indicator (0 to 120 percent of full power).

The isolated nuclear power signals are available for recording by the nuclear instrumentation system recorder. An isolated nuclear power signal is available for recording overpower conditions up to 200% of full power.

Alarm signals for dropped-rod-rod stop, overpower-rod stop, overpower (low and high range)-reactor trips, and channel tests are annunciated on the main control board. Control signals which are sent to the reactor control and protection system include dropped-rod-rod stop, overpower-rod stop, overpower-reactor trip, and permissive circuit signals. These are described in Section 7.2

Over-riding the turbine runback and rod stop signals from a Power Range Nuclear Instrument Dropped Rod circuit in a single channel, or over-riding any turbine runback signal alone has no impact on reactor safety.

Intermediate Range Channels

There are two intermediate range channels which utilize two compensated ionization chambers. Direct current from the ion chambers is transmitted through triaxial cables to transistor logarithmic current amplifiers in the nuclear instrumentation equipment.

The logarithmic amplifier derives a signal proportional to the logarithm of the current as received from the output of the compensated ion chamber. The output of the logarithmic amplifier provides an input to the level bistables for reactor protection purposes and source range cutoff. The bistable trip units are similar to those in the other ranges. The trip outputs can be manually blocked after receiving a permissive signal from the power range channels. On decreasing power, the intermediate range trips for reactor protection are automatically inserted when the power range permissive signal is not present.

Low voltage power supplies contained in each drawer furnish the necessary positive and negative voltages for the channel electronic equipment. Two medium voltage power supplies, one in each channel, furnish compensating voltage to the two compensated ion chambers. The high voltage for the compensated ion chambers is supplied by separate power supplies also located in the intermediate range drawers.

Neutron (log N) flux level indicators are mounted, one each, on the front panel of the intermediate range channel cabinet and on the control board. These indicators are calibrated in terms of ion chamber current (10^{-11} to 10^{-3} amp).

Isolated neutron flux level signals are available for recording and startup rate computation. The startup rate for each channel is indicated at the main control board in terms of decades per minute over the range of –0.5 to 5.0 DPM.

Channel test, high flux level rod stop, and reactor trip signals are alarmed on the main control board annunciator. The latter signal is sent to the Reactor Protection System.

Source Range Channels

There are two source range channels utilizing proportional counter detectors. Neutron flux, as measured in the primary shield area, produces current pluses in the detectors. These preamplified pulses are applied to transistor amplifiers and discriminators located in the racks. Triaxial cable is used for all interconnections from the detector assemblies to the instrumentation in the racks. The preamplifiers are located inside the Reactor Containment.

These channels indicate the source range neutron flux and startup rate. They provide high flux level reactor trip and alarm signals to the Reactor Control and Protection Systems. The reactor trip signal is manually blocked when a permissive signal from the intermediate range is available. These channels are also used at shutdown to provide audible alarms in the Reactor Containment and Control Room of any inadvertent increase in reactivity. An audible count rate signal is used during initial phases of startup and is audible in both the Reactor Containment and Control Room.

Amplifiers are used to obtain a high level signal prior to elimination of noise and gamma pulses by the discriminator. The discriminator output is shaped for use by the log integrator.

The log integrator generates an analog signal proportional to the logarithm of the number of pulses per unit time as received from the output of the previous unit. This unit performs log integration of the pulse rate to determine the count rate, and a linear amplifier amplifies the log integrator output for indication, recording, control, and rate computation through isolation amplifiers.

Each source range channel contains two bistable trip units. Both units trip on high flux level, but one is used during shutdown to alarm reactivity changes and the other provides overpower protection during shutdown and startup. The shutdown alarm unit is blocked manually prior to startup or can serve as a startup alarm. When the input to either unit below its set point, the bistable is in its normal position and assumes a “fully-on” status. When an input from the log amplifier reaches or exceeds the set point, the unit reverses its condition and goes “fully-off.” The output of the reactor trip unit controls relays in the Reactor Protection System.

Power supplies furnish the protective and negative voltages for the transistor circuits, the alarm lights, and the adjustable high voltage for the neutron detector.

A test calibration unit can insert selected test or calibration signals into the preamplifier channel input or the log amplifier input. A set of precalibrated level signals are provided to perform channel tests and calibrations. An alarm is registered on the main control board annunciator whenever a channel is being tested or calibrated. A trip bypass switch is also provided to prevent a reactor trip during channel test under certain reactor conditions.

The neutron detector high-voltage cutoff assembly receives a trip signal when a one-out-of-two matrix, controlled by intermediate range channel flux level bistables, and manual block condition are present. The cutoff assembly disconnects the voltage from the source range channel high voltage power supply to prevent operation of the proportional counter outside its design range. High voltage and reactor trip circuits are reactivated automatically when two of the intermediate range signals are below the permissive trip setting.

Mounted on the front panel of the source range channel is a neutron flux level indicator calibrated in terms of count rate level (10^0 to 10^6 cps). Mounted on the control board is a neutron count rate level indicator (100 to 106 cps). Isolated neutron flux signals are available for recording by the Nuclear Instrumentation System recorder and for startup rate computation. The startup rate for each channel is indicted at the main control board in terms of decades per minute over the range of –0.5 to +5.0 DPM. The isolation network for these signals prevents any electrical malfuncton in the external circuitry from affecting the signal being supplied to the flux level bistables. The signals for the channel test, high neutron flux at shutdown, and source range reactor trip are alarmed on the main control board annunciator.

Excore Neutron Flux Monitoring System

The Excore Neutron Flux Monitoring System consists of two redundant trains, each with a Wide range flux detector, locally mounted amplifier and processor, local indications and dedicated penetration feedthroughs and cabling (see Plant Drawing 9321-LL-96553 [Formerly Figure 7.4-4]). Detector sensitivities are illustrated on Figure 7.4-1).

Each of the detectors are fission chambers consisting of two aluminum electrodes electroplated with uranium, insulators and fill gas all included in a titanium assembly. The detectors are located at the 90' and 270' instrument wells and replace the back-up source range detectors that were originally located there. (See Figure 7.4-3)

The amplifiers and microprocessors are located outside the Containment Building in the electrical penetration area in local panels. Redundant trains are powered by redundant instrument bus power supplies. Through isolation devices the Excore Neutron Flux Monitoring System provides the 10 CFR 50, Appendix R, and Reg. Guide 1.97 required shutdown signal. Although both channels provide local and control room (via QSPDS & Plant Computer) indication, only the detector at the 270' location has the alternate electrical feed capability for Appendix R.

The magnitude of the neutron flux in the reactor core is proportional to the fission power in the reactor. The number of neutron pulses per unit time from the detector is proportional to the magnitude of the neutron flux at the detector and since this magnitude is proportional to the neutron flux in the core, the detector pulse rate is therefore proportional to reactor power.

The number of pulses from the detector is monitored and the mean square value of the variance signal from the detector is measured. This mean square value is proportional to the average rate of neutron pulses. The signal processor takes this signal and processes it into a measure of the logarithm of the countrate, the rate of change of countrate, the logarithm of reactor power and the rate of change of reactor power. It provides analog voltage outputs for each of these signals and also provides the isolated outputs as required.

Auxiliary Equipment

Comparator Channel

The comparator channel compares the four nuclear power signals of the power range channels with one another. A local alarm on the channel is actuated when any two channels deviate from one another by a preset adjustable amount. During full power operation, the comparator serves to sense and annunciate channel failures and/or deviations.

Dropped Rod Protection

As backup to the primary protection for the dropped RCC accident, i.e., the rod bottom signal, independent detection is provided by means of the out-of-core power range nuclear channels. The dropped-rod sensing unit contains a difference amplifier, which compares the instantaneous nuclear power signal with an adjustable power lag signal and responds with a trip signal to the bistable amplifier when the difference exceeds a preset adjustable amount. Above a given power level, the signal blocks automatic rod withdrawal and initiates protective action in the form of a turbine load cutback. No credit is taken in the dropped rod accident analysis for turbine runback.

Audio Count Rate Channel

The auto count rate channel provides audible source range information during refueling operations in both the Control Room and the Reactor Containment. In addition, this channel signal is fed to a scaler-timer assembly which produces a visual display of the count rate for an adjustable sampling period.

Recorders

One large, two-pen strip chart recorder is mounted on the main control board for recording the complete range of the source and intermediate channels. It is also possible to record any two power range channels as linear signals. Variable chart speeds have been provided.

Switching of inputs to the recorders does not cause any spurious signals that would initiate false alarms or reactor trips.

Two two-pen recorders are provided to record the flux level from each of the four nuclear power range quadrants.

Power Supply

The Nuclear Instrumentation System is powered by four 120 volts AC independent vital bus circuits. (See Chapter 8)

7.4.3 System Evaluation

Loss of Power

The nuclear instrumentation draws its primary power from vital instrument buses discussed in Chapter 8.

Loss of nuclear instrumentation power would result in the initiation of all reactor trips associated with the channel power failure. In addition, all trips which were blocked prior to loss would be unblocked and initiated.

Reliability and Redundancy

The requirements established for the reactor protective system apply to the nuclear instrumentation. All channel functions are independent of every other channel.

Safety Factor

The relations of the power range channels to the Reactor Protective System has been described in Section 7.2. To maintain the desired accuracy in trip action, the total error from drift in the power range channels is held to ±1 percent of full power. Routine tests and recalibration ensure that this degree of deviation is not exceeded. Bistable trip set points of the power range channels are also held to an accuracy of ±1 percent of full power. The accuracy and stability of the equipment were verified by vendor tests.

Overpower Trip Set Point

The overpower trip set point for the Indian Point 3 Reactor is 109%. This trip set point was selected to provide adequate assurance that spurious reactor trips would not occur during normal operation. Table 7.4-1 lists the factors which make up the maximum overpower level of 118% based upon a trip set point of 109%.

TABLE 7.4-1

INSTRUMENTATION DRIFT AND CALORIMETRIC ERRORS NUCLEAR OVERPOWER TRIP CHANNEL

	Set Point and Error Allowances: (% of rated power)	Estimated Instrument Errors: (% of rated power)
Nominal Set Point	109	-
Calorimetric Error	2	1.55
Axial power distribution effects on total ion chamber current	5	3
Instrumentation channel drift and set point reproductibility	2	1.0
Maximum overpower trip point assuming all individual errors are simultaneously in the most adverse direction	118	-

7.5 PROCESS INSTRUMENTATION

7.5.1 Design Bases

The non-nuclear process instrumentation measures temperatures, pressures, flows, and levels in the Reactor Coolant System, Steam System, Reactor Containment and Auxiliary Systems. Process variables required on a continuous basis for the startup, operation, and shutdown of the unit are indicated, recorded and controlled from the Control Room. The quantity and types of process instrumentation provided ensure safe and orderly operation of all systems and processes over the full operating range of the plant.

Certain controls which require a minimum of operator attention, or are only in use intermittently, are located on local control panels near the equipment to be controlled. Monitoring of the alarms of such control systems are provided in the Control Room.

Certain process variable indications for normal operation and post accident conditions are made available in the Control Room and the emergency response facilities through the Plant Computer.

7.5.2 System Design

Much of the process instrumentation provide din the plant has been described in the Reactor Control System, the Reactor Protection System and the Nuclear Instrumentation System descriptions (see Sections 7.2, 7.3 and 7.4, respectively). The most important instrumentation used to monitor and control the plant have been described in the above systems descriptions. The remaining portion of the process instrumentation is generally shown on the respective systems process flow diagrams.

Condensate pots and wet legs are used to prevent process temperatures from actually reaching the transmitters.

Reactor Vessel Level Indicating System (RVLIS)

The Reactor Vessel Level Indicating System (RVLIS) provides a means to monitor the water level in the reactor vessel during a postulated accident. It is designed to function under all normal, abnormal, accident and post-accident conditions concurrent with seismic events. The RVLIS consists of two redundant trains, with redundant power supplies, which automatically compensate for variations in fluid density as well as for the effects of reactor coolant pump operation.

The level instrumentation is divided into the full range (Δ_{PF}) and the dynamic range (Δ_{PF}) in order to measure level under all conditions. The full range gives level indication from the bottom of the reactor vessel to the top of the reactor head during natural circulation conditions. The dynamic range gives indication of reactor vessel liquid level for any combination of running RCP's. Comparison of indicated d/p against an algorithm derived ΔP gives a relative void content of the coolant in the core. (See Figure 7.5-2)

The RVLIS utilizes RCS penetrations to manual isolation valves. At the valves are sealed capillary impulse lines (two at the reactor head and two at the seal table) which transmit pressure measurements to d/p transmitters located outside the Containment Building in the in

the Primary Auxiliary Building. The capillary impulse lines are sealed at the RCS end and at the penetrations (inside Containment) with sensor bellows which serve as hydraulic couplers. The impulse lines extend through the Containment wall to hydraulic isolators which seal and isolate the lines as well as provide hydraulic coupling to capillary tubes going to the d/p transmitters. Inside the Containment Building, strap-on RTD's are utilized for vertical runs of impulse lines to correct the reference leg density contributions to the d/p measurement. (See Figure 7.5-2)

Engineered Safety Features

The following instrumentation ensures coverage of the effective operation of the engineered safety features:

Containment Pressure

The containment pressure is transmitted to the main control board for post accident monitoring. Six transmitters, two in each of three safety channels, are installed outside the containment to prevent potential missile damage. The pressure is indicated (all six measurement loops) on the main control board; the range is –5 psig to 75 psig.

The six measurement loops, monitoring containment pressure, reflect the effectiveness of engineered safety features.

Separate from the above, a continuous record of containment pressure is provided in a separate recorder panel in the Control Room. Two redundant and separately channeled safety related, Containment Building pressure measurements are transmitted to and recorded in the Control Room; their range is –5 psig to +200 psig. Each pressure measurement loop consists of a pressure transmitter, a pressure recorder and the necessary signal conditioning equipment, including a power supply, located in the Control Room. Each measurement loop is powered from a separate safety related 118 volts AC instrument bus. (See Section 5.5)

Two local high accuracy, narrow range pressure gauges, capable of directly monitoring containment pressure, are provided in the PAB Fan House, at Elevation 41 ft. These local gauges can be used by Operations, when required, to maintain containment pressure within more restrictive limits, based on RWST and containment temperatures, as defined in Technical Specification 3.6.4, "Containment Pressure". Two pressure gauges are provided for increased reliability only, as these gauges do not perform a safety-related indication function.

Containment Building Hydrogen Concentration

Indication of hydrogen in the Containment Building during and after a postulated accident is available from redundant sample conditioners and analyzers. The concentration is continually recorded by 2 recorders located in the Control Room.

Containment Building and Sump Water Level

There are measuring loops for monitoring water level in the Containment Sump, Recirculation Sump and the Containment Building. Each loop consists of a sensor and transmitter located in the Containment Building and a power supply and recorder in the Control Room.

In addition, to alert the operator in the event of a flooding incident, a reactor pit water level alarm provides indication in the Control Room; and a water level sensing probe and remote control unit provide containment sump overflow indication to the Control Room.

Refueling Water Storage Tank Level

Two redundant channels indicate that Safety Injection and Containment Spray Systems have removed water from the storage tank. One level indication and two low level alarms are transmitted from the tank to the control board.

Safety Injection Pumps Discharge Pressure

These channels show that the safety injection pumps are operating. The transmitters are outside the Containment.

Safety Injection System Flows

Flow indication is provided to the control board for the high and low head injection lines and the recirculation phase containment spray lines.

Pump Energization

All pump motor power feed breakers indicate that they have closed by energizing indicatng lights on the control board.

Valve Position

All engineered safety features valves have position indication on the control board to show proper positioning of the valves. Air operated and solenoid operated valves are selected so as to move in a preferred direction on the loss of air or power. Motor-operated valves remain in the position they held at the time of loss of power to the motor.

Residual Heat Exchangers

Individual exit flows are indicated, plus combined inlet temperature and individual exit temperatures are recorded, on the control board to monitor operation of the residual heat exchangers.

Service Water

Individual service water pump flows are monitored through the use of an annubar flow measurement system. This system provides flow indication at the service water pump location.

Air Coolers

Local flow indication is provided outside containment for service water flow to each cooling unit. Abnormal flow alarms are provided in the Control Room. Service water common inlet temperatures, and all outlet temperatures are displayed on the Plant Computer. A Control Room alarm is actuated if the flow is low coincident with a safety injection signal. The transmitters are outside the Reactor Containment. In addition, the exit flow is monitored for radiation and alarmed in the Control Room if high radiation should occur. This is a common

monitor and the faulty cooler can be located by manually blocking the flow to each unit in turn with locally operated valves.

Alarms

Visual and audible alarms are provided to call attention to abnormal conditions. The alarms are of the individual acknowledgment type; that is, the operator must recognize and silence the audible alarm for each alarm point. For most control systems, the sensing device and circuits for the alarms are independent, or isolated from, the control devices.

In addition to the above, the following local instrumentation is available:

a) Containment spray test lines total flow
b) Safety injection test line pressure and flow

Monitoring Systems

A Safety Parameter Display System (SPDS) is provided to the Control Room which continuously displays information from which plant status can be assessed. Information on the following functions is provided:

a) Reactivity Control
b) Reactor core cooling and heat removal from the primary system
c) Reactor coolant system integrity
d) Radioactivity control
e) Containment conditions

The SPDS consists of the Plant Computer and the Qualified Safety Parameters Display System (QSPDS). The Plant Computer displays and alarms of critical safety functions (set of actions, which preserve integrity of one or more physical barriers against radiation) are indicated in the Control Room (CR) and the three emergency response facilities Technical Support Center (TSC), Emergency Operations Facility (EOF) and Alternate Emergency Operations Facility (AEOF). The Plant Computer is a partially redundant computer system not designed to seismic and electrical class 1E criteria. The QSPDS is a backup display system to the Plant Computer that is qualified to seismic and electrical class 1E standards.

The QSPDS design and display is based on NRC Regulatory Guide 1.97 criteria. The Plant Computer provides for historical data storage and retrieval capability (HDSR). The HDSR system will record, store, recall and display historical information either as graphs and trends or printed logs.

The Plant Computer/QSPDS receive signals from various plant equipment. The Plant Computer receives signals from safety related and non-safety related sources, and adequate electrical separation is maintained by use of fiber optic links.

In order to comply with the requirements of Regulatory Guide 1.97, additions to the original plant design parameters were made. Transmitters monitoring many process variables were installed and the Plant Computer is utilized to alarm and display these parameters. In some cases local indicators are also provided to facilitate local operation needs. Besides additions, replacement of existing components were made to upgrade them to meet the requirements.

7.5.3 System Evaluation

Redundant instrumentation has been provided for all inputs to the protective system and vital control circuits.

Where wide process variable ranges and precise control are required, both wide range and narrow range instrumentation is provided.

All electrical and electronic instrumentation required for safe and reliable operation is supplied from four redundant instrumentation buses.

7.5.4 Instrument Required

Table 7.5-1 identifies the instruments used to demonstrate compliance with NRC Regulatory Guide 1.97. Exemptions to compliance are noted in the table.

The Technical Specifications establish required actions and completion times for Regulatory Guide 1.97 Type A and Category 1 instrument channels.

In addition, inoperability of the following associated recorders is limited to 14 days: Containment Pressure, Containment Water Level, Recirculation Sump Water Level, Containment Hydrogen Monitor [Historical Information], Steam Generator Water level (Wide Range), RCS Pressure (Wide Range), Cold Leg Temperature (Wide Range), Hot Leg Temperature (Wide Range), Pressurizer Water Level, RCS Subcooling Monitor.

Surveillance requirements for Regulatory Guide 1.97 Type A and Category 1 instruments are established in the Technical Specifications. In addition, a Channel Operational Test is required, as follows, for alarms that are associated with Type A and Category 1 instruments, but which have no Regulatory Guide function:

- Main Steam Line Radiation (R62), Quarterly
- Gross Failed Fuel Detector (R63A/B), Quarterly
- [Deleted]
- Containment Hydrogen Monitor, Monthly [Historical Information]

TABLE 7.5-1
Regulatory Guide 1.97 Instruments Required

REG GUIDE 1.97		STATUS OF COMPLIANCE			
INDEX	**TYPE CAT**	**VARIABLE ONE**	**VARIABLE TWO**	**INST LOOP**	**NOTES**
101A	A1	Primary Coolant	Pressure, Reactor Coolant System, Loop 1	P402	J
101B	A1	Primary Coolant	Pressure, Reactor Coolant System, Loop 4	P403	J
102A	A1	Primary Coolant	Temperature, Hot Leg Loop No. 1	T413A	P
102B	A1	Primary Coolant	Temperature, Hot Leg Loop No. 2	T423A	P
102C	A1	Primary Coolant	Temperature, Hot Leg Loop No. 3	T433A	P
102D	A1	Primary Coolant	Temperature, Hot Leg Loop No. 4	T443A	P
103A	A1	Primary Coolant	Temperature, Cold Leg Loop No. 1	T413B	P
103B	A1	Primary Coolant	Temperature, Cold Leg Loop No. 2	T423B	P
103C	A1	Primary Coolant	Temperature, Cold Leg Loop No. 3	T433B	P
103D	A1	Primary Coolant	Temperature, Cold Leg Loop No. 4	T443B	P
104A	A1	Steam Generator 31	Level, Wide Range	L417D	K
104B	A1	Steam Generator 31	Level, Narrow Range	L417A	K
Deleted					
104D	A1	Steam Generator 31	Level, Narrow Range	L417C	K
104E	A1	Steam Generator 32	Level, Wide Range	L427D	K
104F	A1	Steam Generator 32	Level, Narrow Range	L427A	K
Deleted					
104H	A1	Steam Generator 32	Level, Narrow Range	L427C	K
104I	A1	Steam Generator 33	Level, Wide Range	L437D	K
104J	A1	Steam Generator 33	Level, Narrow Range	L437A	K
Deleted					
104L	A1	Steam Generator 33	Level, Narrow Range	L437C	K
104M	A1	Steam Generator 34	Level, Wide Range	L447D	K
104N	A1	Steam Generator 34	Level, Narrow Range	L447A	K
Deleted					
104P	A1	Steam Generator 34	Level, Narrow Range	L447C	K
105A	A1	Pressurizer	Level, Channel I	L459	
105B	A1	Pressurizer	Level, Channel II	L460	
105C	A1	Pressurizer	Level, Channel III	L461	

TABLE 7.5-1
Regulatory Guide 1.97 Instruments Required

REG GUIDE 1.97		STATUS OF COMPLIANCE			
INDEX	**TYPE CAT**	**VARIABLE ONE**	**VARIABLE TWO**	**INST LOOP**	**NOTES**
106B	A1	Containment	Wide Range Pressure, Channel I	P1421	O
106C	A1	Containment	Wide Range Pressure, Channel II	P1422	O
107A	A1	Steam Generator 31	Pressure, Channel I	P419A	
107B	A1	Steam Generator 31	Pressure, Channel II	P419B	
107C	A1	Steam Generator 31	Pressure, Channel IV	P419C	
107D	A1	Steam Generator 32	Pressure, Channel I	P429A	
107E	A1	Steam Generator 32	Pressure, Channel II	P429B	
107F	A1	Steam Generator 32	Pressure, Channel IV	P429C	
107G	A1	Steam Generator 33	Pressure, Channel I	P439A	
107H	A1	Steam Generator 33	Pressure, Channel II	P439B	
107I	A1	Steam Generator 33	Pressure, Channel IV	P439C	
107J	A1	Steam Generator 34	Pressure, Channel I	P449A	
107K	A1	Steam Generator 34	Pressure, Channel II	P449B	
107L	A1	Steam Generator 34	Pressure, Channel IV	P449C	
108A	A1	Refueling Water Storage Tank	Level, Alarm	L920	N
108B	A1	Refueling Water Storage Tank	Level, Alarm	L921 / L923	N
109A	A1	Containment	Water Level	L1253	L
109B	A1	Containment	Water Level	L1254	L
111A	A1	Containment	Radiation, Area, High Range	R25	
111B	A1	Containment	Radiation, Area, High Range	R26	
112A	A1	Secondary Cooling	Radiation, Main Steam	R62	SS
113A	A1	Primary Coolant	Temperature, Core Exit	CE-T-***	TT
114A	A1	Condensate Storage Tank Level	Water Level	L1128	
114B	A1	Condensate Storage Tank Level	Water Level	L1128A	
115A	A1	Primary Coolant	Temperature, Degrees of RCS Subcooling	QSPDS-A	M

TABLE 7.5-1
Regulatory Guide 1.97 Instruments Required

REG GUIDE 1.97		STATUS OF COMPLIANCE			
INDEX	**TYPE CAT**	**VARIABLE ONE**	**VARIABLE TWO**	**INST LOOP**	**NOTES**
115B	A1	Primary Coolant	Temperature, Degrees of RCS Subcooling	QSPDS-B	M
201A	B1	Neutron Flux Excore	Radiation, Intermediate Range Channel I	N38	
201B	B1	Neutron Flux Excore	Radiation, Intermediate Range Channel II	N39	
202A	B3	Control Rods	Position	N/A	
203A	B3	Primary Coolant	Sampling, Soluble Boron Concentration	N/A	Grab Sample
204A	B3	Primary Coolant	Temperature, Cold Leg, Loop No. 1	T413B	P
204B	B3	Primary Coolant	Temperature, Cold Leg, Loop No. 2	T423B	P
204C	B3	Primary Coolant	Temperature, Cold Leg, Loop No. 3	T433B	P
204D	B3	Primary Coolant	Temperature, Cold Leg. Loop No. 4	T433B	P
205A	B1	Primary Coolant	Temperature, Hot Leg, Loop No. 1	T413A	P
205B	B1	Primary Coolant	Temperature, Hot Leg, Loop No. 2	T423A	P
205C	B1	Primary Coolant	Temperature, Hot Leg, Loop No. 3	T433A	P
205D	B1	Primary Coolant	Temperature, Hot Leg, Loop No. 4	T443A	P
206A	B1	Primary Coolant	Temperature, Cold Leg, Loop No. 1	T413B	P
206B	B1	Primary Coolant	Temperature, Cold Leg, Loop No. 2	T423B	P
206C	B1	Primary Coolant	Temperature, Cold Leg, Loop No. 3	T433B	P
206D	B1	Primary Coolant	Temperature, Cold Leg, Loop No. 4	T443B	P
207A	B1	Primary Coolant	Pressure, Reactor Coolant System, Loop 1	P402	J
207B	B1	Primary Coolant	Pressure, Reactor Coolant System, Loop 4	P403	J
208A	B3	Primary Coolant	Temperature, Core Exit	CE-T-***	TT
209A	B1	Primary Coolant	Level, Reactor	RVLIS TR-A & B	
210A	B2	Primary Coolant	Temperature, Degrees of Subcooling	QSPDS-A	
210B	B2	Primary Coolant	Temperature, Degrees of Subcooling	QSPDS-B	
211A	B1	Primary Coolant	Pressure, Reactor Coolant System, Loop 1	P402	J
211B	B1	Primary Coolant	Pressure, Reactor Coolant System, Loop 4	P403	J
212C	B2	Containment	Level, Containment Sump Water Channel I	L1255	L
212D	B2	Containment	Level, Containment Sump Water Channel II	L1256	L
212E	B1	Containment	Level, Wide Range Channel I	L1253	L

TABLE 7.5-1
Regulatory Guide 1.97 Instruments Required

REG GUIDE 1.97		STATUS OF COMPLIANCE			
INDEX	**TYPE CAT**	**VARIABLE ONE**	**VARIABLE TWO**	**INST LOOP**	**NOTES**
212F	B1	Containment	Level, Wide Range Channel II	L1254	L
212I	B2	Containment	Level, Wide Range Redundant Channel: Recirculation Sump Level-Channel I	L1251	L
212J	B2	Containment	Level, Wide Range Redundant Channel: Recirculation Sump Level-Channel II	L1252	L
213B	B1	Containment	Pressure, Channel I	P1421	O
213C	B1	Containment	Pressure, Channel II	P1422	O
214A	B1	Containment	Position, Isolation valve	N/A	Y
215B	B1	Containment	Pressure, Channel I	P1421	O
215C	B1	Containment	Pressure, Channel II	P1422	O
301A	C1	Primary Coolant	Temperature, Core Exit	CE-T-***	TT
302A	C1	Primary Coolant	Radiation, Radioactivity Concentration	R-63A&B	
303A	C1	Primary Coolant	Radiation, Gamma Spectrum	N/A	W
304A	C1	Primary Coolant	Pressure, Reactor Coolant System Loop 4	P402	J
304B	C1	Primary Coolant	Pressure, Reactor Coolant System Loop 1	P403	J
305B	C1	Containment	Pressure, Channel I	P1421	O
305C	C1	Containment	Pressure, Channel II	P1422	O
306C	C2	Containment	Level, Containment Sump Water Channel I	L1255	L
306D	C2	Containment	Level, Containment Sump Water Channel II	L1256	L
306E	C1	Containment	Level, Wide Range Channel I	L1253	L
306F	C1	Containment	Level, Wide Range Channel II	L1254	L
306I	C1	Containment	Level, Wide Range Redundant Channel: Recirculation Sump Level-Channel I	L1251	L
306J	C1	Containment	Level, Wide Range Redundant Channel: Recirculation Sump Level-Channel II	L1252	L
307A	C3	Containment	Radiation, Area	R25	
307B	C3	Containment	Radiation, Area	R26	
308A	C3	Cond Air Removal Sys Exhaust	Radiation, Effluent Noble Gas	R15	

TABLE 7.5-1
Regulatory Guide 1.97 Instruments Required

REG GUIDE 1.97		STATUS OF COMPLIANCE			
INDEX	TYPE CAT	VARIABLE ONE	VARIABLE TWO	INST LOOP	NOTES
309A	C1	Primary Coolant	Pressure, Reactor Coolant System, Loop 1	P402	J
309B	C1	Primary Coolant	Pressure, Reactor Coolant System, Loop 4	P403	J
310B	C3	Containment Air	Sampling, Hydrogen Concentration Channel I	HCMC-A	
310C	C3	Containment Air	Sampling, Hydrogen Concentration Channel II	HCMC-B	
311B	C1	Containment	Pressure, Channel I	P1421	O
311C	C1	Containment	Pressure, Channel II	P1422	O
312A	C2	Containment	Radiation, Effluent, Noble Gas, Penetration Area	R12	AA
314B	C2	Penetration Area	Radiation, Area, Electrical Tunnel In Area of Electrical Penetration	N/A	BB
314C	C2	Penetration Area	Radiation, Area, 83' Personnel Airlock Area	N/A	BB
314D	C2	Penetration Area	Radiation, Area, Containment Purge Valve Area Between Containment & Fan House	N/A	BB
314E	C2	Penetration Area	Radiation, Area, 95' Personnel & Equipment Hatch Area	N/A	BB
314F	C2	Penetration Area	Radiation, Area, Fuel Transfer Area Between Containment & Fuel Storage Buildings	N/A	BB
314G	C2	Fuel Storage Building	Radiation, Area, Penetration Area, In Area of Fuel Transfer Tube	R5	BB
314H	C2	PAB 34' FL EL	Radiation Area, Piping Tunnel In Area of Containment Sump Drain Pent	N/A	BB
314J	C2	PAB 54' FL EL	Radiation, Area, Piping Tunnel in Area of Piping Penetrations	N/A	BB
401A	D2	Residual Heat Removal	Flow Rate, Header 31	F638	
401B	D2	Residual Heat Removal	Flow Rate, Header 32	F640	
401C	D2	Residual Heat Removal	Flow Rate, Loop 4	FT946A	
401D	D2	Residual Heat Removal	Flow Rate, Loop 3	FT946B	
401E	D2	Residual Heat Removal	Flow Rate, Loop 2	FT946C	
401F	D2	Residual Heat Removal	Flow Rate, Loop 1	FT946D	
402A	D2	Residual Heat Removal	Temperature, Heat Exchanger 31 Outlet	T639	

TABLE 7.5-1
Regulatory Guide 1.97 Instruments Required

REG GUIDE 1.97		STATUS OF COMPLIANCE			
INDEX	**TYPE CAT**	**VARIABLE ONE**	**VARIABLE TWO**	**INST LOOP**	**NOTES**
402B	D2	Residual Heat Removal	Temperature, Heat Exchanger 32 Outlet	T641	
403A	D2	Safety Injection	Level, Accumulator Tank 31	L934A	Z
304B	D2	Safety Injection	Level, Accumulator Tank 32	L934B	Z
403C	D2	Safety Injection	Level, Accumulator Tank 33	L934C	Z
403D	D2	Safety Injection	Level, Accumulator Tank 34	L934D	Z
403E	D2	Safety Injection	Pressure, Accumulator Tank 31	P937A	Z
403F	D2	Safety Injection	Pressure, Accumulator Tank 32	P937B	Z
403G	D2	Safety Injection	Pressure, Accumulator Tank 33	P937C	Z
403H	D2	Safety Injection	Pressure, Accumulator Tank 34	P937D	Z
404A	D2	Safety Injection	Pressure, Accumulator Tank 31 Isolation Valve 894A	N/A	HH
404B	D2	Safety Injection	Pressure, Accumulator Tank 32 Isolation Valve 894B	N/A	HH
404C	D2	Safety Injection	Pressure, Accumulator Tank 33 Isolation Valve 894C	N/A	HH
404D	D2	Safety Injection	Pressure, Accumulator Tank 34 Isolation Valve 894D	N/A	HH
405A	D2	Safety Injection	Flow, Boric Acid Charging	F128	H
406A	D2	Safety Injection	Flow, High Head, Cold Leg Loop I	F926	
406B	D2	Safety Injection	Flow, High Head, Cold Leg Loop I	F924A	
406C	D2	Safety Injection	Flow, High Head, Cold Leg Loop 2	F981	
406D	D2	Safety Injection	Flow, High Speed, Cold Leg Loop 2	F925	
406E	D2	Safety Injection	Flow, High Speed, Cold Leg Loop 3	F980	
406F	D2	Safety Injection	Flow, High Speed, Cold Leg Loop 3	F926A	
406G	D2	Safety Injection	Flow, High Speed, Cold Leg Loop 4	F982	
406H	D2	Safety Injection	Flow, High Speed, Cold Leg Loop 4	F927	
407A	D2	Safety Injection	Flow, Low Head	F638	
407B	D2	Safety Injection	Flow, Low Head	F640	
408A	D2	Safety Injection	Level, Refueling Water Storage Tank	L920	

TABLE 7.5-1
Regulatory Guide 1.97 Instruments Required

REG GUIDE 1.97		STATUS OF COMPLIANCE			
INDEX	**TYPE CAT**	**VARIABLE ONE**	**VARIABLE TWO**	**INST LOOP**	**NOTES**
409A	D3	Primary Coolant	Status, Reactor Coolant Pump 31	N/A	
409B	D3	Primary Coolant	Status, Reactor Coolant Pump 32	N/A	
409C	D3	Primary Coolant	Status, Reactor Coolant Pump 33	N/A	
409D	D3	Primary Coolant	Status, Reactor Coolant Pump 34	N/A	
410A	D2	Primary Coolant	Position, Safety Relief Valve, Power Operated Relief Valve 455C	N/A	Acoustical Monitor At Valve UU
410B	D2	Primary Coolant	Position, Safety Relief Valve, Power Operated Relief Valve 456	N/A	Acoustical Monitor At Valve
410C	D2	Primary Coolant	Position, Safety Relief Valve, ASME Code Safety Valve 464	N/A	Acoustical Monitor At Valve
410D	D2	Primary Coolant	Position, Safety Relief Valve, ASME Code Safety Valve 466	N/A	Acoustical Monitor At Valve
410E	D2	Primary Coolant	Position, Safety Relief Valve, ASME Code Safety Valve 468	N/A	Acoustical Monitor At Valve
411A	D1	Primary Coolant	Level, Pressurizer Channel I	L459	
411B	D1	Primary Coolant	Level, Pressurizer Channel II	L460	
411C	D1	Primary Coolant	Level, Pressurizer Channel III	L461	
412A	D2	Primary Coolant	Status, Pressurizer Heater – Control Group	N/A	U
412B	D2	Primary Coolant	Status, Pressurizer Heater – Back-up Group 31	N/A	U
412C	D2	Primary Coolant	Status, Pressurizer Heater – Back-up Group 32	N/A	U
412D	D2	Primary Coolant	Status, Pressurizer Heater – Back-up Group 33	N/A	U
413A	D3	Primary Coolant	Level, Pressurizer Relief Tank 31	L470	
414A	D3	Primary Coolant	Temperature, Pressurizer Relief Tank 31	T471	
415A	D3	Primary Coolant	Pressure, Pressurizer Relief Tank 31	P472	

TABLE 7.5-1
Regulatory Guide 1.97 Instruments Required

REG GUIDE 1.97		STATUS OF COMPLIANCE			
INDEX	**TYPE CAT**	**VARIABLE ONE**	**VARIABLE TWO**	**INST LOOP**	**NOTES**
416A	D1	Secondary Cooling	Level, Steam Generator 31	L417D	K
416B	D1	Secondary Cooling	Level, Steam Generator 32	L427D	K
416C	D1	Secondary Cooling	Level, Steam Generator 33	L437D	K
416D	D1	Secondary Cooling	Level, Steam Generator 34	L447D	K
417A	D2	Secondary Cooling	Pressure, Steam Generator 31, Channel I	P419A	
417B	D2	Secondary Cooling	Pressure, Steam Generator 32, Channel I	P429A	
417C	D2	Secondary Cooling	Pressure, Steam Generator 33, Channel I	P439A	
417D	D2	Secondary Cooling	Pressure, Steam Generator 34, Channel I	P449A	
418A	D2	Secondary Cooling	Flow, Main Steam From Steam Generator 31	F419A&B	
418B	D2	Secondary Cooling	Flow, Main Steam From Steam Generator 32	F429A&B	
418C	D2	Secondary Cooling	Flow, Main steam From Steam Generator 33	F439A&B	
418D	D2	Secondary Cooling	Flow, Main Steam From Steam Generator 34	F449A&B	
419A	D3	Secondary Cooling	Flow, Main Feedwater To Steam Generator 31	F418A&B	
419B	D3	Secondary Cooling	Flow, Main Feedwater To Steam Generator 32	F428A&B	
419C	D3	Secondary Cooling	Flow, Main Feedwater To Steam Generator 33	F438A&B	
419D	D3	Secondary Cooling	Flow, Main Feedwater To Steam Generator 34	F448A&B	
420A	D2	Secondary Cooling	Flow, Auxiliary Feedwater To Steam Generator 31	F1200R	
420B	D2	Secondary Cooling	Flow, Auxiliary Feedwater To Steam Generator 32	F1201R	
420C	D2	Secondary Cooling	Flow, Auxiliary Feedwater To Steam Generator 33	F1202R	
420D	D2	Secondary Cooling	Flow, Auxiliary Feedwater To Steam Generator 34	F1203R	
421A	D1	Secondary Cooling	Level, Condensate Storage Tank Water	L1128	G
421B	D1	Secondary Cooling	Level, Condensate Storage Tank Water	L1128A	G
422A	D2	Containment	Flow, Spray From Residual Heat Removal Heat Exchanger 31	F945B	II
422B	D2	Containment	Flow, Spray From Residual Heat Removal Heat	F945A	II

TABLE 7.5-1
Regulatory Guide 1.97 Instruments Required

REG GUIDE 1.97		STATUS OF COMPLIANCE			
INDEX	**TYPE CAT**	**VARIABLE ONE**	**VARIABLE TWO**	**INST LOOP**	**NOTES**
			Exchanger 32		
423A	D2	Containment	Flow, Heat Removal By System-Service Water RCFC 31	F1121	
423B	D2	Containment	Flow, Heat Removal By system-Service Water RCFC 32	F1122	
423C	D2	Containment	Flow, Heat Removal By System-Service Water RCFC 33	F1123	
423D	D2	Containment	Flow, Heat Removal By System-Service Water RECF 34	F1124	
423E	D2	Containment	Flow, Heat Removal By System-Service Water RCFC 35	F1125	
423F	D2	Containment	Temperature, Heat Removal By System-Service Water Diff RCFC 31	T-1415-1	
423G	D2	Containment	Temperature, Heat Removal By System-Service Water Diff RCFC 32	T-1415-2	
423H	D2	Containment	Temperature, Heat Removal By System-Service Water Diff RCFC 33	T-1415-3	
423J	D2	Containment	Temperature, Heat Removal By-System-Service Water Diff RCFC 34	T-1415-4	
423K	D2	Containment	Temperature, Heat Removal By System-Service Water Diff RCFC 35	T-1415-5	
424A	D2	Containment	Temperature, Atmosphere	T1203	
425A	D2	Containment	Temperature, Sump Water	NONE	I
426A	D2	Chemical & Volume Control	Flow, Make-up In	F128	
427A	D2	Chemical & Volume Control	Flow, Letdown Out	F134	B
428A	D2	Chemical & Volume Control	Level, Volume Control Tank	L112	C

TABLE 7.5-1
Regulatory Guide 1.97 Instruments Required

REG GUIDE 1.97		STATUS OF COMPLIANCE			
INDEX	**TYPE CAT**	**VARIABLE ONE**	**VARIABLE TWO**	**INST LOOP**	**NOTES**
429A	D2	Component Cooling	Temperature, Component Cooling Heat Exchanger 31 Output	T602A	D
429B	D2	Component Cooling	Temperature, Component Cooling Heat Exchanger 32 Output	T602B	D
430A	D2	Component Cooling	Flow, Component Cooling Heat Exchanger 31 Output	F601A	E
430B	D2	Component Cooling	Flow, Component Cooling Heat Exchanger 32 Output	F601B	E
431A	D3	Radwaste	Level, High-Level Radioactive Waste Hold-up Tank 31	L1001	
431B	D3	Radwaste	Level, High-Level Radioactive Waste Hold-up Tank 32 (3HBT01A)	L168	JJ
431C	D3	Radwaste	Level, High-Level Radioactive Waste Hold-up Tank 33 (3HBT01B)	L170	JJ
432A	D3	Radwaste	Pressure, Large Radioactive Gas Decay Tank 31	P1036	KK
432B	D3	Radwaste	Pressure, Large Radioactive Gas Decay Tank 32	P1037	KK
432C	D3	Radwaste	Pressure, Large Radioactive Gas Decay Tank 33	P1038	KK
432D	D3	Radwaste	Pressure, Large Radioactive Gas Decay Tank 34	P1039	KK
432E	D3	Radwaste	Pressure, Small Radioactive Gas Decay Tank 31	P1052	KK
432F	D3	Radwaste	Pressure, Small Radioactive Gas Decay Tank 32	P1053	KK
432G	D3	Radwaste	Pressure, Small Radioactive Gas Decay Tank 33	P1054	KK
432H	D3	Radwaste	Pressure, Small Radioactive Gas Decay Tank 34	P1055	KK
432J	D3	Radwaste	Pressure, Small Radioactive Gas Decay Tank 35	P1056	KK
432K	D3	Radwaste	Pressure, Small Radioactive Gas Decay Tank 36	P1057	KK
433A	D2	Ventilation	Position, Reactor Containment Fan Cooler 31 Damper A & B	N/A	GG
433B	D2	Ventilation	Position, Reactor Containment Fan Cooler 31 Damper A & B	N/A	GG
433C	D2	Ventilation	Position, Reactor Containment Fan Cooler 31	N/A	GG

TABLE 7.5-1
Regulatory Guide 1.97 Instruments Required

REG GUIDE 1.97		STATUS OF COMPLIANCE			
INDEX	**TYPE CAT**	**VARIABLE ONE**	**VARIABLE TWO**	**INST LOOP**	**NOTES**
			Damper D & Blow-in Door		
433D	D2	Ventilation	Position, Reactor Containment Fan Cooler 32 Damper A & B	N/A	GG
433E	D2	Ventilation	Position, Containment Fan Cooler 32 Damper C	N/A	GG
433F	D2	Ventilation	Position, Reactor Containment Fan Cooler 32 Damper D & Blow-in Door	N/A	GG
433G	D2	Ventilation	Position, Reactor Containment Fan Cooler 33 Damper A & B	N/A	GG
433H	D2	Ventilation	Position, Reactor Containment Fan Cooler 33 Damper C	N/A	GG
433J	D2	Ventilation	Position, Reactor Containment Fan Cooler 33 Damper D & Blow-in Door	N/A	GG
433K	D2	Ventilation	Position, Reactor Containment Fan Cooler 34 Damper A & B	N/A	GG
433L	D2	Ventilation	Position, Reactor Containment Fan Cooler 34 Damper C	N/A	GG
433M	D2	Ventilation	Position, Reactor Containment Fan cooler 34 Damper D & Blow-in Door	N/A	GG
433N	D2	Ventilation	Position, Reactor Containment Fan Cooler 35 Damper A & B	N/A	GG
433P	D2	Ventilation	Position, Reactor Containment Fan Cooler 35 Damper C	N/A	GG
433R	D2	Ventilation	Position, Reactor Containment Fan Cooler 35 Damper D & Blow-in Door	N/A	GG
433S	D2	Ventilation	Position, Fuel Storage Building Forced Air Unit 31 Emergency Damper	N/A	GG
433T	D2	Ventilation	Position, Fuel Storage Building Forced Air Unit 32 Emergency Damper	N/A	GG
433U	D2	Ventilation	Position, Fuel Storage Building Normal Airflow Top Damper	N/A	GG

TABLE 7.5-1
Regulatory Guide 1.97 Instruments Required

REG GUIDE 1.97		STATUS OF COMPLIANCE			
INDEX	**TYPE CAT**	**VARIABLE ONE**	**VARIABLE TWO**	**INST LOOP**	**NOTES**
433V	D2	Ventilation	Position, Fuel Storage Building Normal Airflow Bottom Damper	N/A	GG
433W	D2	Ventilation	Position, Fuel Storage Building Emergency Airflow Filter Intake Damper	N/A	GG
433X	D2	Ventilation	Position, Fuel Storage Building Emergency Airflow Filter Exhaust Damper	N/A	GG
433Y	D2	Ventilation	Position, Primary Auxiliary Building Exhaust Charcoal Damper – Face	N/A	GG
433Z	D2	Ventilation	Position, Primary Auxiliary Building Exhaust Charcoal Damper – Bypass	N/A	GG
434A	D2	Emergency Power	Current, AC Bus 31	N/A	
434B	D2	Emergency Power	Current, AC Bus 32	N/A	
434C	D2	Emergency Power	Current, AC Bus 33	N/A	
434D	D2	Emergency Power	Current, AC Bus 34	N/A	
434E	D2	Emergency Power	Voltage, AC Bus 31	N/A	
434F	D2	Emergency Power	Voltage, AC Bus 32	N/A	
434G	D2	Emergency Power	Voltage, AC Bus 33	N/A	
434H	D2	Emergency Power	Voltage, AC Bus 34	N/A	
434I	D2	Emergency Power	Current, DC Bus 31	N/A	F
434J	D2	Emergency Power	Current, DC Bus 32	N/A	F
434K	D2	Emergency Power	Current, DC Bus 33	N/A	F
434L	D2	Emergency Power	Current, DC Bus 34	N/A	F
434M	D2	Emergency Power	Voltage, DC Bus 31	N/A	
434N	D2	Emergency Power	Voltage, DC Bus 32	N/A	
434O	D2	Emergency Power	Voltage, DC Bus 33	N/A	
434P	D2	Emergency Power	Voltage, DC Bus 34	N/A	
434Q	D2	Emergency Power	Current, Diesel 31	N/A	
434R	D2	Emergency Power	Current, Diesel 32	N/A	
434S	D2	Emergency Power	Current, Diesel 33	N/A	

TABLE 7.5-1
Regulatory Guide 1.97 Instruments Required

REG GUIDE 1.97		STATUS OF COMPLIANCE			
INDEX	**TYPE CAT**	**VARIABLE ONE**	**VARIABLE TWO**	**INST LOOP**	**NOTES**
434T	D2	Emergency Power	Voltage, Diesel 31	N/A	
434U	D2	Emergency Power	Voltage, Diesel 32	N/A	
434V	D2	Emergency Power	Voltage, Diesel 33	N/A	
434W	D2	Emergency Air Supply	Pressure, Instrument Air Receiver Tank	P1207	
434X	D2	Emergency Air Supply	Pressure, Diesel 31 Starting Air Receiver Tank	N/A	
434Y	D2	Emergency Air Supply	Pressure, Diesel 32 Starting Air Receiver Tank	N/A	
434Z	D2	Emergency Air Supply	Pressure, Diesel 33 Starting Air Receiver Tank	N/A	
501A	E1	Containment	Radiation, Area, High Range	R25	
501B	E1	Containment	Radiation, Area, High Range	R26	
502A	E3	Central Control Room	Radiation, Area	R1	MM,CC,X
502B	E3	PAB 80'	Radiation, Area, Charging Pump Room	R4	DD
502C	E3	Fuel Storage Building	Radiation, Area	R5	
502D	E3	PAB 55'	Radiation, Area, Sampling Room (North Wall)	R6	X
502E	E2	Containment	Radiation, Area, (AT Seal Table) In-core Instrument Room	R7	X, DD
502F	E2	PAB 55'	Radiation, Area, Drumming Station	R8	X, DD
502G	E2	Aux Boiler Feed Pump Bldg	Radiation, Area, (West Wall Opposite Main Steam Penetrations 31 & 32)	NONE	X
502H	E2	PAB 55'	Radiation, Area, On Column Across From Sample Room	R64	
502J	E2	PAB 73'	Radiation, Area, Entrance Way To Volume Control Tank	N/A	X
502K	E2	PAB 73'	Radiation, Area, Hall Next To NPO Office	R65	
502L	E2	PAB 41'	Radiation, Area, South Wall Area Of Refueling Water Purification Pumps	N/A	X
502M	E2	PAB 41'	Radiation, Area, Hall On Column Next To Containment Spray Pumps	N/A	X
502N	E2	PAB 34'	Radiation, Area, Hall Near Entry To Safety Injection Pumps	R66	

TABLE 7.5-1
Regulatory Guide 1.97 Instruments Required

REG GUIDE 1.97		STATUS OF COMPLIANCE			
INDEX	**TYPE CAT**	**VARIABLE ONE**	**VARIABLE TWO**	**INST LOOP**	**NOTES**
502P	E2	PAB 41'	Radiation, Area, Pipe Tunnel In Area Of Chemistry Post Accident Sampling Station	R67	
502Q	E2	PAB 15'	Radiation, Area, On North Wall Adjacent To RHR Valve Gallery	R68	
502R	E2	RAB 15'	Radiation, Area, Hall On Wall At Entry To Filter Cell	N/A	X
502S	E2	PAB 54'	Radiation, Area, Within The Doorway On The Wall, Pipe Penetration	R69	
502T	E2	PAB 67'	Radiation, Area, Above Pipe Penn In Area Of Hydrogen Recombiner Panels	N/A	X
502U	E2	Fan Building 92'	Radiation, Area, In Area Of 4 Channel Iodine Monitors	R70	
502V	E2	Fan Building 72'	Radiation, Area, Outside Plenum In Area Of Differential Pressure Instruments	R70	
503A	E2	Containment	Radiation, Effluent, Noble Gas	R27	Via Plant Vent
504A	E2	Reactor Shield Building Annulus	Radiation, Effluent, Noble Gas	N/A	
505A	E2	Auxiliary Building	Radiation, Effluent, Noble Gas, Or Others Containing Primary System Gases	R27	Via Plant Vent
506A	E2	Cond Air Removal Sys Exhaust	Radiation, Effluent, Noble Gas	R15	NN
506B	E2	Cond Air Removal Sys Exhaust	Radiation, Effluent, Noble Gas – Flow Rate	R15	
507 A	E2	Common Plant Vent	Radiation, Effluent, Noble Gas	R27	SS
507B	E2	Common Plant Vent	Radiation, Effluent, Flow Rate	R27	SS
508A	E2	Steam Generator	Radiation, Effluent, Noble Gas From Safety Relief Valves Or Atm Dump Valves	R62	FF
509A	E2	Admin Bldg Exhaust Vent	Radiation, Effluent, Noble Gas From 4th Floor	R46	OO, CC

TABLE 7.5-1
Regulatory Guide 1.97 Instruments Required

REG GUIDE 1.97		STATUS OF COMPLIANCE			
INDEX	TYPE CAT	VARIABLE ONE	VARIABLE TWO	INST LOOP	NOTES
509B	E2	Admin Bldg Exhaust Vent	Radiation, Effluent, Flow Rate, 4th Floor	NONE	OO
509C	E2	Radioactive Machine Shop Exhaust Vent	Radiation, Effluent, Noble Gas	R59	
509D	E2	Radioactive Machine Shop Exhaust Vent	Radiation, Effluent, Flow Rate	FT-1776	
509E	E2	Steam Generator Blowdown	Radiation, Effluent	R19	
509F	E2	Steam Generator Blowdown	Radiation, Effluent, Flow Rate	F538	
510A	E3	Common Plant Vent	Radiation, Effluent, Particulates	N/A	EE, SS
510B	E3	Common Plant Vent	Radiation,, Effluent, Halogens	N/A	EE, SS
510C	E3	Common Plant Vent	Radiation, Effluent, Flow Rate	R27	
510D	E3	Admin Bldg Exhaust Vent	Radiation, Effluent, Particulates From The 4th Floor	N/A	DD, OO
510E	E3	Admin Bldg Exhaust Vent	Radiation, Effluent, Halogens From The 4th Floor	N/A	DD, OO
510F	E3	Admin Bldg Exhaust Vent	Radiation, Effluent, Flow Rate, 4th Floor	NONE	DD, OO
510G	E3	Radioactive Machine Shop Exhaust Vent	Radiation, Effluent, Particulates	N/A	CC
510H	E3	Radioactive machine shop exhaust vent	Radiation, Effluent, Halogens	NONE	CC
510J	E3	Radioactive machine shop exhaust vent	Radiation, Effluent, Flow Rate	FT-1776	
511A	E3	Environs	Radiation, Exposure Rate	N/A	RR
512A	E3	Environs	Radiation, airborne radiohalogens and particulates	N/A	portable instrum.
513A	E3	Environs	Radiation, photons	N/A	portable instrum.
513B	E3	Environs	Radiation, beta and low energy photons	N/A	portable instrum.

TABLE 7.5-1
Regulatory Guide 1.97 Instruments Required

REG GUIDE 1.97		STATUS OF COMPLIANCE			
INDEX	**TYPE CAT**	**VARIABLE ONE**	**VARIABLE TWO**	**INST LOOP**	**NOTES**
514A	E3	Environs	Radioactivity, multi channel gamma-ray spectrometer	N/A	
515A	E3	Meteorological	Met, wind direction	N/A	
516A	E3	Meteorological	Met, wind speed	N/A	
517A	E3	Meteorological	Met, atmospheric stability	N/A	
518A	E3	Sampling	Primary coolant and containment sump water analysis – gross activity	N/A	W,R
518B	E3	Sampling	Primary coolant and containment sump water analysis – gamma spectrum	N/A	W,R
518C	E3	Sampling	Primary coolant and containment sump water analysis – boron content	N/A	W,R

Table 7.5-1

Regulatory Guide 1.97 Instruments Required
NOTES

General Notes that apply to all items have an * as an identifier.

NOTE A: DELETED

NOTE B: The letdown flow is controlled by opening a remote operated valve, which allows flow through fixed orifice plates. The maximum CVCS letdown flow allowed administratively is limited to 120 gpm. It is the Authority's position that the indicated range (0-125 gpm) is adequate.

NOTE C: The existing level (18% to 82%) transmitter range is adequate. The modification necessary to obtain the additional level (0%-100%) required by 1.97 is not warranted based on manrem exposure and cost versus benefit.

NOTE D: The existing range indication for component cooling heat exchanger temperature is adequate for all modes of normal operation of off-normal modes of operation. The temperature of the component cooling system to date has not decreased below the existing range of 50°F. In addition, in the event of a major accident the temperature would be expected to increase as opposed to decrease, further assuring that the temperature would not decrease below the low range of the temperature system.

NOTE E: The existing range indication for component cooling heat exchanger flow is adequate for all modes of normal operation or off-normal modes of operation. The component cooling flow indication during normal operation may decrease below the existing range however; this condition does not cause any concern warranting a modification. The pump can be assured that it is functioning via low pressure and pump breaker status alarms. The components that are being cooled have local flow devices that are used to regulate the flow; therefore, minimum pump flow conditions can be met. In addition, in the event of a major accident, the flow would increase as opposed to decrease.

NOTE F: It is the Authority's position that sufficient indication to D.C. bus status is provided to the operators such that during post accident conditions, the operators will be aware of the operability of the D.C. buses.

NOTE G: Condensate storage tank level is currently monitored by two-(2) independent qualified transmitters. Diverse indication of CST level can be derived by auxiliary feedwater suction pressure indication. It is the Authority's position that the existing monitoring of CST level complies with the requirements of Regulatory Guide 1.97.

NOTE H: Boric acid flow to the RCS is monitored by the high-pressure injection (HPI) flow transmitters. Refer to index number 406 A-H which meets Reg. Guide 1.97 requirements.

NOTE I: Based on conversations with the NRC staff, the intent of this variable may be satisfied by the indication of several other variables. IP-3 has indication of RHR outlet temperature, containment spray flow and containment temperature which provides adequate indication of containment heat removal capability.

NOTE J: Adequate diverse measurement to PT-402 and PT-403 is obtained from pressure transmitters used to monitor pressurizer pressure (PT-455, 456, 457 and 474) for the range of 1700-2500 psig. Additionally, R.C.S. pressure, 0-3000 psig is indicated on a pressure gauge located in an area accessible to plant operators.

NOTE K: Each Steam Generator contains four (4) transmitters to indicate steam generator water level Three (3) transmitters per steam generator indicate narrow range level which is a span that begins at the top of the tube bundles to the moisture separator. Two of these three (3) Narrow Range Transmitters (L4X7A & L4X7C) are credited with providing R.G. 1.97 isolation. The remaining level transmitter covers the span from the bottom tube sheet up to the moisture separator. Based on above, diversity exists from the top portion of the steam generator. Two (2) auxiliary feedwater flow indicators provide a diverse indication for the steam generator. In addition, since two of our four steam generators are required for heat removal, redundant wide range level for each generator is deemed not necessary.

NOTE L: Two (2) redundant level transmitters (LT-1253 & 1254) provide containment water level indication to the Central Control Room (CCR) operators. In addition, the containment sump and recirculation sump each contain (2) qualified level transmitters. The refueling water storage tank provides a diverse measurement for the containment water level.

NOTE M: Diversity is met via a third system which records saturation pressure margin and also use of steam tables.

NOTE N: Containment water level provides a diverse method to determine refueling water storage tank level.

NOTE O: Additional Containment pressure instrumentation exists (PT 948A, B & C and PT 949A, B & C) to provide a diverse means of establishing containment pressure.

NOTE P: Redundancy for the Hot Leg Reactor Coolant Temperature will be by the use of the core exit thermocouples (Diverse Variable). Redundancy for the Cold Leg Reactor Coolant Temperature is provided by the steamline pressure instrument PT 419 A, B & C; PT 429 A, B, & C; PT 439 A, B, & C and PT 449 A, B, & C (Diverse Variable).

NOTE Q: DELETED

NOTE R: DELETED

NOTE S*: On March 4, 1983, the NRC conducted a workshop in Chicago, Illinois in order to clarify the technical requirement of NUREG-0737, Supplement I. The handout distributed by the NRC at this workshop states that with respect to seismic qualification requirement for operating reactors, it will suffice to state that instrumentation systems comply with the

seismic qualification program which was the basis for plant licensing. Accordingly, the seismic requirement is indicated in Enclosure B as being satisfied if that instrumentation complies with the licensing basis for seismic qualification. [GENERAL NOTE]

NOTE T*: As noted in Regulatory Guide 1.97, Revision 3, Category 1 and 2 instrumentation should be qualified in accordance with Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants," and the methodology described in NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." Enclosure B reflects this requirement for all Category 1 and 2 instrumentation. However, certain Category 1 and 2 instrumentation are located in mild post-accident environments and therefore are not within the scope of Regulatory Guide 1.89. For the sake of convenience, the Category 1 and 2 instrumentation located in a mild post-accident environment are noted as meeting Environmental Qualification (E.Q.) requirement. Hence, that instrumentation noted in Enclosure B as satisfying the E.Q. requirement either satisfy the requirements of 10 CFR 50.49 or are located in a mild post-accident environment.

NOTE U: Since the purpose of Pressurizer Heater Status is to ensure that they do not overload a diesel, adequate diesel generator loading information is available to the operators. The heaters are supplied by a safety related electrical bus and are stripped from that bus in the event of a Safety Injection Signal. They must be manually placed in service by the control room operator and procedures are in place that provide the guidance to ensure the diesels are not overloaded.

In addition, heater electrical breaker status lights are available. The pressurizer pressure and temperature response also provides verification that the heaters are operational.

NOTE V: DELETED

NOTE W: The Authority concurred with the NRC approach to post-accident sampling capability review. The deviations are beyond the scope of the Regulatory Guide 1.97 submittal and are best addressed via our submittal to NuReg-0737, Item II.B.3

NOTE X: Portable survey meters are the primary source of data on the radiation exposure rates inside buildings. These portable instruments are used to 1) verify the indication of the existing installed radiation monitors, and 2) determine exposure rates where there are no installed radiation monitors. It is Entergy's opinion that the portable survey meters meet the intent of the Guide.

NOTE Y: The automatic containment isolation valves at the facility meet all of the requirements of the Regulatory Guide on position indication. Non-automatic containment isolation valves are not provided with position indication. Valves that are considered essential and non-automatic are maintained in the open position and are closed after the initial phases of an accident. Approved emergency procedures are utilized to control the closing of these valves. Non-essential

containment isolation valves are maintained in the closed position and may be opened, if necessary, for plant operation and for only as long as necessary to perform the intended function, as required by Indian Point 3 Technical Specifications. These valves are additionally administratively controlled in the following manner:

1. Shift Manager approval for opening a non-automatic containment isolation valve is required.
2. An operator must be dedicated to the operation of these valves as long as they are in the open position.
3. Operator must have communications established with the Central Control Room, and
4. Operators first response to any emergency condition while the valve is open is to insure that the valve is returned to the closed position.

NOTE Z: Since the accumulators will discharge immediately when RCS pressure drops below accumulation pressure, these variables are unnecessary following an accident. Since power to the isolation valves is locked out at the circuit breaker, the operator would not be able to utilize these variables for manual actions, except for events in which the RCS pressure is decreasing very slowly. For such events, the present indicators are expected to function properly. Letter from NRC (N. F. Conicella) to R. Beedle, dated 9/28/92, entitled “REGULATORY GUIDE 1.97 – INSTRUMENTATION TO FOLLOW THE COURSE OF AN ACCIDENT FOR INDIAN POINT GENERATING UNIT NO. 3 (TAC No. M51099)”, relaxed the requirement for Accumulator Pressure and Level Instrumentation and deleted the commitment for upgrading Accumulator Pressure and Level Instrumentation.

NOTE AA: The original radiation monitor used to monitor containment effluent radioactivity (R-12) is located in a non-harsh environmental area. Therefore, the environmental qualification requirements of the regulatory guide are satisfied. The combination of R-12 and an additional environmentally qualified effluent radiation monitor (R-27) sufficiently meets the range requirements of the Regulatory Guide.

NOTE BB: Radiation exposure rates inside buildings or areas in direct contact with primary containment where penetrations and hatches are located can be sufficiently monitored by portable radiation monitoring detectors.

NOTE CC: The existing sampler or radiation monitors for these areas do not meet the range requirements of the Regulatory Guide, however, it is Entergy’s position that the indicated range is sufficient for the highest levels that are postulated for these areas.

NOTE DD: The existing area radiation monitors for these areas do not meet the range requirements of the Regulatory Guide, however, it is Entergy’s position that these areas need not be monitored for the mitigation of an accident.

NOTE EE: To accommodate the range requirements of these radiation detectors, Entergy will use the Post Accident Sampling System.

NOTE FF: The plant computer will record the steam release duration and mass flow rate.

NOTE GG: Damper indication status is provided via red-green indicating lamps in the control room. The lamps are illuminated by a single limit switch, which is toggled when the damper is in the opened or closed position.

The Containment Fan Cooler units are provided with flow switches, which will cause an annunciation in the control room if low flow exists. In addition, a Weir system exists to quantify the cooling and condensing features of the ventilation unit.

Since failure of dampers are rare and it is improbable that the limit switch or some diverse variable would not detect the failure, it is Entergy's position that no modifications are warranted.

NOTE HH: The white lights used to satisfy Index 404A, B, C, and D are on when the valves are fully open and off when not fully open. These lights are always operable.

The valves are opened and the power and control circuits are de-energized when the RCS pressure is above 1000 psi. When these circuits are energized, each valve has red and green indicator lights which tell the operator whether the valve is full open, full closed or at some intermediate position.

NOTE II: The containment spray system consists of 4 spray headers. Two headers are used during the initial phase of the accident and the other two headers are used later in the accident. Manual operator action based on spray system flow rates is required in the later phase of the accident. As such, the spray flow indications described in Enclosure B are provided by the two headers used later in the accident only.

NOTE JJ: The existing level represents approximately 94% of the tank range. Since the tanks are horizontal cylindrical, the level actually monitors greater than 94% of its volume. These tanks are back up to 31 Waste Hold-Up tanks.

NOTE KK: The range that is required by the Guide, 0 to 165 psig, exceeds the tank design pressure and the tank safety valve setting, i.e., 150 psig. As additional status of tank pressure, an alarm is actuated when tank pressure reaches 110 psig. It is therefore concluded that the actual range of tank pressure is acceptable and meets the intent of the Regulatory Guide.

NOTE LL: DELETED – Monitor R-10 has been removed from the plant.

NOTE MM: The control room monitor's range is considered adequate. The operators would evacuate the control room prior to fields reaching the upper range prescribed in Reg. Guide 1.97.

NOTE NN: This possible atmospheric release point is designed to divert into the containment at relatively low levels. In addition, prior to reaching 1.97 levels, you would have to have fuel damage, steam generator tube failures and failure of the diversion to containment feature, which are highly improbable. Main steam radiation monitors are capable of detecting activity that would escape from condenser air ejectors. It is Entergy's position that the existing monitor is adequate to monitor the release point.

NOTE OO: The monitor is located and provides radiation level in an area that is not considered part of the plant proper. No radioactivity materials are expected to be brought into this area that would warrant any increase in the range of the existing monitors or the addition of flow monitoring devices.

NOTE PP*: As per Regulatory Guide 1.97 Rev. 3, seismic qualification is not required for Category II variables. [GENERAL NOTE]

NOTE QQ: DELETED

NOTE RR: No longer required as per Rev. 3 of Regulatory Guide 1.97.

NOTE SS: If the plant vent sampling capability, the wide-range vent monitor, or the main steam line radiation monitor is inoperable in MODES 1, 2, or 3, initiate a preplanned alternate sampling / monitoring capability as soon as practical, but no later than 72 hours after identification of the failure.

NOTE TT: The present list of qualified Core Exit Thermocouples is:
K-11, L-12, [Deleted], C-12, F-12, E-10, D-9, A-11, B-3, B-6, E-5, F-5, G-4, R-10, P-13, K-3, J-7, N-2, H-8 & L-1

NOTE UU: Acoustic Monitoring for Relief Valve 455C is non-functional for operating cycle 19. Temperature Element (TE-463), PRT Level (LT-470) or Temperature Element (TE-471) is used for PORV (RC-PCV-455C) Position Indication for operating cycle 19.

7.6 IN-CORE INSTRUMENTATION

7.6.1 Design Basis

The in-core instrumentation is designed to yield information on the neutron flux distribution and fuel assembly outlet temperatures at selected core locations. Using the information obtained from the in-core instrumentation system, it is possible to confirm the reactor core design parameters and calculated hot channel factors. The system provides means for acquiring data and performs no operational plant control.

7.6.2 System Design

The in-core instrumentation system consists of thermocouples, positioned to measure fuel assembly coolant outlet temperature at preselected locations; and flux thimbles, which run the length of selected fuel assemblies to measure the neutron flux distribution within the reactor core. The flux thimble at core location E-11 was retired during 3R17 and permanently removed during 3R18.

The data obtained from the in-core temperature and flux distribution instrumentation system, in conjunction with previously determined analytical information, can be used to determine the fission power distribution in the core at any time throughout core life. This method is more accurate than using calculational techniques alone. Once the fission power distribution has been established, the thermal power distribution and the thermal and hydraulic limitations determine the core capability and maximum power output.

The in-core instrumentation provides information which may be used to calculate the coolant enthalpy distribution, the fuel burnup distribution, and an estimate of the coolant flow distribution.

Both radial and azimuthal symmetry of power may be evaluated by comparing the detector information from quadrant to quadrant.

Thermocouples

Chromel-alumel thermocouples are passed through into guide tubes that penetrate the reactor vessel head through seal assemblies, and terminate at the exit flow end of the fuel assemblies. The thermocouples are provided with two primary seals, a conoseal and swage type seal from conduit to head. The thermocouples are enclosed in stainless steel sheaths within the above tubes to allow replacement if necessary. Thermocouple readings are obtainable via the plant computer and at a manually selected display unit in the control room. The support of the thermocouple guide tubes in the upper core support assembly is described in Chapter 3.

Moveable Miniature Neutron Flux Detectors

Mechanical Configuration

Six fission chamber detectors (employing U_3O_8, which is 93 percent enriched in U_{235}) can be remotely positioned in retractable guide thimbles to provide flux mapping of the core. Maximum chamber dimensions are 0.188-inch in diameter and 2.10 inches in length. The stainless steel detector shell is welded to the leading end of the helical wrap drive cable and the stainless steel sheathed coaxial cable. Each detector is designed to have a minimum thermal neutron

sensitivity of 1.5 x 10^{-17} amps/nv and a maximum gamma sensitivity of 3 x 10^{-14} amps/R/hr. Maximum thermal neutron flux for these detectors is 5 x 10^{13} nv. Other miniature detectors, such as gamma ionization chambers and boron-lined neutron detectors, can also be used in the system. The basic system for the insertion of these detectors is shown in Figures 7.6-2 to 7.6-4. Retractable thimbles into which the miniature detectors are driven are pushed into the reactor core through conduits which extend from the bottom of the reactor vessel down through the concrete shield area and then up to a thimble seal zone.

The thimbles will be closed at the leading ends, are dry inside, and serve as the pressure barrier between the reactor water pressure and the atmosphere. Mechanical seals provided on the retractable thimbles and on the conduits are shown on Figure 7.6-4.

During reactor operation, the retractable thimbles are stationary. They are extracted downward from the core during refueling to avoid interference within the core. A space above the seal line is provided for the retraction operation.

The drive system for the insertion of the miniature detectors consists basically of six drive assemblies, six 5-path rotary group selector assemblies and six 10-path rotary selector assemblies, as shown in Figures 7.6-2 and 7.6-3. The drive system pushes hollow helical-wrap drive cables into the core with the miniature detectors attached to the leading ends of the cables and small diameter sheathed coaxial cables threaded through the hollow centers back to the ends of the drive cables. Each drive assembly generally consists of a gear motor which pushes a helical-wrap drive cable and detector through a selective thimble path by means of a special drive box and includes a storage device that accommodates the total drive cable length. Further information on mechanical design and support is described in Chapter 3.

Control and Readout Description

The control and readout system provides means for inserting the miniature neutron detectors into the reactor core and withdrawing the detectors at a selected speed while plotting a level of induced radioactivity versus detector position. The control system consists of two sections, one physically mounted with the drive units, and the other contained in the control room. Limit switches in each path provide feedback of path selection operation. Each gear box drives an encoder for position feedback. One 5-path group selector is provided for each drive unit to route the detector into one of the flux thimble groups. A 10-path rotary transfer assembly is a transfer device that is used to route a detector into any one of up to ten selectable paths. Manually operated isolation valves allow free passage of the detector and drive wire when open, and prevents steam leakage from the core in case of a thimble rupture, when closed. A common path is provided to permit cross calibration of the detectors.

The control room contains the necessary equipment for control, position indication, and flux recording. Panels are provided to indicate the core position of the detectors, and for plotting the flux level versus the detector position. Additional panels are provided for such features as drive motor controls, core path selector switches, plotting and gain controls. A “flux-mapping” consists, briefly, of selecting (by panel switches) flux thimbles in given fuel assemblies at various core quadrant locations. The detectors are driven or inserted to the top of the core and stopped automatically. A x-y plot (position vs. flux level) is initiated with the slow withdrawal of the detectors through the core from top to a point below the bottom. In a similar manner other core locations are selected and plotted.

The system that will be used to monitor the distribution of power in the X-Y plane is described in WCAP-7669, “Topical Report – Nuclear Instrumentation System.”

Operational limits due to a quadrant power tilt are given in the Technical Specifications.

The calibration of the Nuclear Instrumentation System by the movable incore detector system is made in accordance with the Technical Specifications. As noted in the Technical Specifications, the movable incore detector system shall be used to confirm power distribution.

After the excore system is calibrated initially, recalibration is performed periodically to compensate for changes in the core, due for example to fuel depletion, and for changes in the detectors.

If the recalibration is not performed, the mandated power reduction assures safe operation of the reactor as it will compensate for an error of 10% in the excore protection system. Experience at Beznau No. 1 and R. E. Ginna plants has shown that drift due to changes in the core or instrument channels is very slight. Thus the 10% reduction is considered to be very conservative.

The reactor trip functions (Section 7.2) provide core protection at the safety limits prescribed in the Technical Specifications. Those trip functions derived from the Nuclear Instrumentation System are described in WCAP-7669.

Each detector provides axial flux distribution data along the center of a fuel assembly. Various radial positions of detectors are then compared to obtain a flux map for a region of the core

7.6.3 System Evaluation

The thimbles are distributed throughout the core as shown in Figure 7.6-1. The positions have been chosen to provide symmetry checks and sufficient coverage, taking symmetry into account, to construct a full core three-dimensional power shape. With this number and location of thimbles the measurement accuracy for the peak to average rod in an x-y plane is 3.65% and for the peak to average pellet, including axial peaking, is 4.58%. These accuracies include the flux thimble to hot rod calculational uncertainty and instrumentation repeatability. They represent a 95% confidence level in a probability of fewer than 5% of cases lying above this error allowance. This confidence level and accuracy is consistent with the interpretation of DNB criteria.

The derivation and justification of these uncertainties is given in WCAP-7308-L, “Evaluation of Nuclear Hot Channel Factor Uncertainties.”

7.6.4 System Operation

A. A minimum of 2 thimbles per quardrant and sufficient movable incore detectors shall be operable during recalibration of the excore axial offset detection system.

B. During the incore / excore calibration procedure, full core flux maps will be made only when at least 38 of the movable detector guide thimbles are operable.

7.7 OPERATING CONTROL STATIONS

7.7.1 Station Layout

The principal criteria of control station design and layout is that all controls, instrumentation displays and alarms required for the safe operation and shutdown of the plant are readily available to the operators in the Control Room.

During other than normal operating conditions, other operators will be available to assist the operators in the Control Room. Plant Drawings 9321-F-30523 and -33833 [Formerly Figure 7.7-1 and 7.7-2] show the Control Room layout and sections for the unit. The control board is divided into relative areas to show the location of control components and information display pertaining to various subsystems.

7.7.2 Information Display and Recording

Alarms and annunciators in the Control Room provide the warning to the operators of abnormal plant conditions which might lead to damage of components, fuel or other unsafe conditions. Other displays and recorders are provided for indication of routing plant operating conditions and for the maintenance of records.

Consideration is given to the fact that certain systems normally require more attention from the operator. The control system is therefore centrally located on the three section board.

On the left section of the control board, individual indicators present a direct, continuous readout of every control rod position. Fault detectors in the rod drive control system are used to alert the operator should an abnormal condition exist for any individual or group of control rods. Displayed in this same area are limit lights for each control rod group and all nuclear instrumentation information required to start up and operate the reactor. Control rods are manipulated from the left section.

Subsequent to periods of rod motion, when thermal equilibrium is being established in the rod position indicator coil stacks, temporary drifting of the indicators can be expected. During such time if indicated RCCA position differs from bank demand more than allowed by the Technical Specifications, the rod is treated as potentially misaligned under Technical Specification 3.1.4. Rod position is confirmed via a digital voltage meter applied to the rod position control racks. In addition, the operators will continue to monitor the affected rod position indicators on the main control board (and on the plant computer, if available and in agreement with the digital voltage meter reading) to check for increased deviation.

Variables associated with operation of the secondary side of the station are displayed and controlled from the control board. These variables include steam pressure and temperature, feedwater flow, electrical load, and other signals involved in the plant control system. The control board also contains provisions for indication and control of the reactor coolant system. Redundant indication is incorporated in the system design since pressure and temperature variables of the Reactor Coolant System are used to initiate safety features. Control and display equipment for station auxiliary systems is also located here.

The Engineered Safety Features Systems are controlled and monitored from a vertical panel to the left of the control board. Valve position indicating lights are provided as a means of verifying the proper operation of the control and isolation valves following initiation of the engineered safety features. Control switches located on this panel allow manual operation or test of individual units.

Also located on this section are the control switches, indicating lights, and meters for fans and pumps required for emergency conditions. Also mounted on this section are auxiliary electrical system controls required for manual switching between the various power sources described in Section 8.2.2.

Controls and indications for all ventilation systems, the containment isolation valves, and the Isolation Valves Seal Water System are located on a vertical panel. Radiation monitoring information is indicated immediately behind and to the left of the main control board.

Audible Reactor Building alarms are initiated from the radiation monitoring system and from the source range nuclear instrumentation. Audible alarms will be sounded in appropriate areas throughout the station if high radiation conditions are present.

7.7.3 Emergency Shutdown Control

The Control Room, its equipment and furnishings were designed so that the likelihood of fire or other conditions which could render the Control Room inaccessible even for a short time is extremely small. For details on the fire protection features, refer to Section 9.6.2.

A criterion of the station design and layout was that all controls, instrumentation displays and alarms required for the safe operation and shutdown of the plant are readily available to the operators in the Control Room.

It was design policy that the functional capacity of the Control Room should be maintained at all times inclusive of accident conditions, such as a Maximum Credible Accident or a fire; the following features were incorporated in the design to ensure that this criterion was met.

Structural and finish materials for the Control Room and the cable spreading room below were selected on the basis of fire resistant characteristics. Structural floors are concrete reinforced. Interior partitions are metal paneling joints. The Control Room ceiling covering is fire retardant egg crate diffusers. Door frames and doors are metallic. Wooden trim is not used.

The Control Room is equipped with portable fire extinguishers sized and located in accordance with National Fire Code and National Fire Protection Association specifications. Extinguishers carry the Underwriter's Laboratory label of approval and are electrical shock resistant.

Fire protection features of the cable spreading room and safe shutdown capability in the event of a fire in the cable spreading room are discussed in Section 9.6.2.

The Control Room ventilation consists of a system having a large percentage of recirculated air. The fresh air intake can be closed to control the intake of airborne activity if monitors indicate that such action is appropriate. Two independent control room toxic gas monitors are provided to alert the operators in the event that toxic gases exceed the short-term exposure limit (STEL).

Control cables used throughout the installation have been selected on the basis of flame testing described in Chapter 8 and have superior flame retardant capability. In addition, electrical circuits in the Control Room are limited to those associated with lighting, instrumentation and control. Lighting circuits operate on 120 volts, instrumentation and control circuits operate at

either 120 volt AC, 125 volt DC or at millivolt level. All 120 and 125 volt circuits are protected against both overload and short circuits by either fuses or circuit breakers. The power levels on the millivolt circuits are so low that it is inconceivable that short circuits in these could become a fire hazard.

No process fluids, combustible or otherwise, are carried into the Control Room.

Cables that penetrate the Control Room floor pass through sealing devices to minimize fume and flame transmission from possible fire sources external to the Control Room.

All internal wiring in switchboards and instrument racks has excellent resistance to propagation of flame. As a result of the design criterion discussed above the amount of combustible material in the Control Room is of such small quantity that a fire of the magnitude that would require evacuation of the Control Room is not credible.

As a further measure to assure safety, provisions have been made so that plant operators can shut down and maintain the plant in a safe condition by means of controls located outside the Control Room. During such a period of Control Room inaccessibility the reactor will be tripped and the plant maintained in a hot shutdown condition. If the period extends for a long time, the Reactor Coolant System can be borated to maintain shutdown as xenon decays.

Local controls are located so that the stations to be manned and the times when attention is needed are within the capability of the plant operating staff. The plant intercom system and other communication equipment provide for a flow of information among the personnel so that operation of the facility can be coordinated.

The functions for which local control provisions have been made are listed below along with the type of control and its location in the plant. Transfer to these local controls is annunciated in the Control Room.

<u>Reactor Trip</u>

If the Control Room should be evacuated suddenly without any action by the operators, the reactor can be tripped by any of the following actions:

1) Open rod control breakers in the control building
2) Actuate the manual turbine trip at the control standard in the turbine building, only if above P-8 setpoint 35%
3) Manually trip the rod drive Motor-Generator set in the Control Building

Following evacuation of the Control Room, the following systems and equipment are provided to maintain the plant in a safe shutdown condition from outside the Control Room:

a) Residual heat removal
b) Reactivity control, i.e., boron injection to compensate for fission product decay
c) Pressurizer pressure and level control
d) Electrical System as required to supply the above systems
e) Other equipment, as described

a) <u>Residual Heat Removal</u>

Following a normal plant shutdown, an automatic steam dump control system bypasses steam to the condenser and maintains the reactor coolant temperature at its no load value. This implies the continued operation of the steam dump system, condensate circuit, condenser cooling water, feedwater pumps and steam generator instrumentation. Failure to maintain water supply to the steam generators would result in steam generator dry out after some 34 minutes and loss of the secondary system for decay heat removal. Redundancy and full protection where necessary is built into the system to ensure the continued operation of the steam generators. If the automatic steam dump control system is not available, independently controlled relief valves downstream of each steam generator maintain the steam pressure. These relief valves are further backed up by code safety valves downstream of each steam generator. Numerous calculations, verified by start-up tests, have shown that with the steam generator safety valves operating alone the Reactor Coolant System maintains itself close to the nominal no load condition. The steam relief capability is adequately protected by redundancy and local protection.

For decay heat removal it is only necessary to maintain control on one steam generator.

For the continued use of the steam generators for decay heat removal, it is necessary to provide a source of water, a means of delivering that water and, finally, instrumentation for pressure and level indication.

The normal source of water supply is the secondary feedwater circuit. This implies satisfactory operation of the condenser, air ejector, condenser cooling circuit, etc. In addition to the normal feedwater circuit the plant may fall back on:

1) The condensate storage tanks
2) The city water storage tank
3) The city water supply

Feedwater may be supplied to the steam generators by the two electrical auxiliary feedwater pumps or by the steam driven auxiliary feedwater pump. These pumps and associated valves have local controls.

b) <u>Reactivity Control</u>

Following a normal plant shutdown to hot shutdown condition soluble poison is added to the primary system to maintain sub-critically. For boron addition the Chemical and Volume Control System is used. Routine boration requires the use of:

Changing pumps and volume control tank with associated piping. Boric Acid transfer pumps with tanks and associated piping. Letdown station, non-regenerative heat exchanger and associated equipment, Component Cooling and Service Water Systems. Compressed air for valve operation – manual could be adopted if necessary.

It is worthy of note that with the reactor held at hot shutdown conditions, boration of the plant is not required immediately after shutdown. The xenon transient does not decay to the equilibrium level until at least 9 hours after shutdown and a further period would

elapse before the reactivity shutdown margin provided by the full-length control rods had been canceled. This delay would provide useful time for emergency measures.

c) Pressurizer Pressure and Level Control

Following a reactor trip, the primary temperature will automatically reduce to the no-load temperature condition as dictated by the steam generator temperature, reducing the primary water volume and, if continued pressure control is to be maintained, primary water makeup is required.

The pressurizer level is controlled in normal circumstances by the Chemical and Volume Control System. This requirement implies the charging pump duty referred to for boration plus a guaranteed borated water supply. The facility for boration is provided as described above; it is only necessary to supply water for makeup. Water may readily be obtained from normal sources, i.e., the volume control tank.

Startup of Other Equipment

Although not directly related to plant operation, certain ultimate heat sink safety analyses assume the air temperature inside containment is kept below 130°F. For this reason, the containment air recirculation fan coolers should continuously be in operation. If they have stopped, at least one should be restarted within five minutes, with the others started later as required. Similarly, the nuclear service water pumps will be checked and at least one of them restarted if none are already operating. The fan coolers and the service water pump remote controls are located in the switchgear room.

Electrical Systems

Offsite or onsite emergency power must be available to supply the above systems and equipment for the hot shutdown condition.

Indication and Controls Provided Outside the Control Room

The specific indication and controls provided outside the Control Room for the above capability are summarized as follows:

Indication

1) Level Indication for the Individual Steam Generators
One set visible from the auxiliary feedwater pumps
One set visible from the main feedwater control valves

2) Pressure Indication for the Individual Steam Generators
One set visible from power operated atmospheric dump valve control stations.
One set visible from the auxiliary feedwater pumps

3) Pressurizer Level and Pressure Indicators
One set visible from the auxiliary feedwater pumps
One set visible from the charging pump local control point

4) RCS Temperature Indication
   Loop #31 Thot and Tcold visible from the auxiliary feedwater pumps

5) RCS Flux Indication
   Source range visible from the charging pump local control point

6) RCS Pressure
   One set visible from the auxiliary feedwater pumps
   One set visible from the charging pump local control point

   All instruments at the auxiliary feedwater pumps are grouped on a local gauge board.

Alternate Power Supplies

Alternate Power Supplies have been provided for the following:

1) Component Cooling Water Pump 32
2) Charging Pump 31 or Charging Pump 32
3) Containment Safe shutdown instrument isolation cabinet

The alternate power supplies for items 1, 2, and 3 consist of manual transfer switches located near its respective load that can transfer the load from its normal power supply to the alternate power source – motor control center 312A located in the turbine building. Operation of the manual transfer switches to alternate power will give an annunciator alarm in the control room. Backup Service Water Pump 38 was removed from its normal supply (Bus 3A) and placed on MCC 312A as the normal supply.

Controls

Local stop/start pushbutton motor controls with a selector switch are provided at each of the motors for the equipment listed below. The selector switch will transfer control of the switchgear from the Control Room to local at the motor. Placing the local selector switch in the local operating position will give an annunciator alarm in the Control Room and will turn out the motor control position lights on the Control Room panel. The equipment consists of:

1) The Motor Driven Auxiliary Feedwater Pumps
2) The Charging Pumps
3) The Boric Acid Transfer Pumps

Remote stop/start pushbutton motor controls with a selector switch are provided for each of the motors for the equipment listed below. These controls are grouped at one point in the switchgear room convenient for operation. The selector switch will transfer control of the switchgear form the Control Room to the remote point. Placing the selector switch to local operation will give an annunciator alarm in the Control Room and will turn out the motor control position lights on the Control Room panel. The equipment consists of:

1) The Service Water Pumps 31 thru 36
2) The Containment Air Recirculation Fans
3) The Control Room Air Handling Unit Including Control for the Air Inlet Dampers

Key operated control switches located on MCC 312A provided local control for:

1) Component Cooling Water Pump 32
2) Charging Pump 31 or Charging Pump 32
3) Backup Service Water Pump 38

when utilizing the alternate power capabilities for items 1 and 2. Alternate motor control points are not required for the following:

1) The Component Cooling Water Pumps (Automatically restarted on a blackout once the diesel generators are operating)

2) The instrument Air Compressors and Cooling Pumps (These will start automatically on low pressures in the air and water services once the diesel automatically energizes the bus and the motor control centers are manually energized. The control point is local to the compressors.)

Isolation Switch Cabinets

Switching cabinets have been provided to permit local operation of Diesel Generator No. 31, its associated 480V load centers and to permit local indication of containment safe shutdown instrumentation (steam generator level, pressurizer level, RCS loop 31 temperature, RCS pressure and pressurizer pressure), independent of the effects of a cable spreading room fire. See Section 9.6.2.4.

Speed Control

Speed control is provided locally for:

1) The turbine Driven Auxiliary Feedwater Pump
2) The Charging Pump

Valve Control

Local valve control is provided at:

1) The Main Feed Regulators

2) The Auxiliary Feed Control Valves (These valves are located local to the auxiliary feedwater pumps)

3) The Atmospheric Dump (Auto control normally at hot shutdown)

4) All other valves requiring operation during hot standby

5) The Letdown orifice isolation valves locally to the charging pumps Local stop and start buttons with selector switch and position lamp are provided.

Pressurizer Heater Control

Stop and start buttons with selector switch and position lamp locally to the charging pumps for one 555 kW backup heater group are provided.

Lighting

Emergency lighting is provided in all operating areas as defined by the foregoing.

Communications

The communication network provides communications between the area of the auxiliary feedwater pumps and the charging pumps, boric acid transfer pumps, diesel generators, and the outside exchange without requiring the Control Room.

7.7.4 Cold Shutdown from Outside the Control Room

Hot shutdown is a stable plant condition, automatically reached following a plant shutdown. The hot shutdown condition can be maintained for an extended period of time. In the unlikely event that access to the Control Room is restricted, the plant can be safely kept at hot shutdown until the Control Room can be re-entered by the use of the monitoring indicators and the controls listed in Section 7.7.3. It is noted that these indicators and controls are provided outside as well as inside the Control Room.

By the use of appropriate equipment and procedures, the reactor can be brought to a cold shutdown condition from locations outside the Control Room if occupancy of the main Control Room should become untenable. The equipment systems that can be made available for a cold shutdown are as follows:

a) Auxiliary feedwater pumps
b) Boric acid transfer pumps
c) Charging pumps
d) Service water pumps
e) Containment fans
f) Component cooling pumps
g) Residual heat removal pumps
h) Controlled steam release equipment (e.g., steam dump valve) and feedwater supply
i) Equipment furnishing a boration capability
j) Safety injection pumps
k) Nuclear Instrumentation:
   1) Excore neutron flux detector channel associated with App R alternate capability
   2) Alternate Power supply if instrumentation power is lost
l) Reactor coolant inventory control equipment (Charging and letdown)
m) Pressurizer, pressure control equipment (heater and spray) including opening control for pressurizer relief valves
n) Certain motor control center and switchgear sections which supply power to the above equipment

In addition, the safety injection signal trip circuit must be defeated and the accumulator isolation valves closed.

Detailed procedures to be followed in achieving cold shutdown from outside the Control Room are best determined by plant personnel at the time of a postulated incident. This is because an assessment of plant conditions can be made on a long term basis (a week or more) to establish procedures for making the necessary physical modifications to instrumentation and control equipment in order to attain a cold shutdown. During such time, the plant could be safely maintained at hot shutdown condition. The reactor plant design does not preclude attaining the cold shutdown condition from outside the Control Room.

7.8 MAXIMUM SAFETY SYSTEM SETTINGS AND MINIMUM CONDITIONS FOR OPERATION

Table 7.2-1 lists the reactor protection and engineered safety features actuation systems and Table 7.2-2 lists the associated interlocks. Maximum permissible settings for safe operation for these functions are given in the Technical Specifications.

7.9 SURVEILLANCE REQUIREMENTS

The requirements for periodic testing of instruments are listed in the Technical Specifications, Technical Requirements Manual, the FSAR, and the ODCM. The type of test action (i.e., channel calibration, channel operational test, etc.) to be taken and the minimum testing frequency (i.e., 31 days, 92 days, 24 months, etc.) for the indicated instruments are provided within the above-mentioned documents.

As indicated, the instrumentation channels which are covered include, for example, nuclear, reactor coolant temperature and flow, pressurizer pressure and level, and auxiliary process channels; or components necessary to assure that facility operation is maintained within the safe limits.