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3/4.3.3.2 (This Specification number is not used.)

TABLE 3.3-9

REMOTE SHUTDOWN PANEL MONITORING INSTRUMENTATION

	INSTRUMENTS*	MINIMUM CHANNELS OPERABLE
1.	Intermediate Range Nuclear Flux	1
2.	Intermediate Range Startup Rate	1
3.	Source Range Nuclear Flux	1
4.	Source Range Startup Rate	1
5.	Reactor Coolant Temperature - Hot Leg	1
6.	Reactor Coolant Temperature - Cold Leg	1
7.	Pressurizer Pressure	1
8.	Pressurizer Level	1
9.	Steam Generator Pressure	1/steam generator
10.	Steam Generator Level	1/steam generator
11.	RHR Temperature - HX Outlet	1
12.	Auxiliary Feedwater Flow Rate	1/steam generator

*Emergency Shutdown Panel

3/4.3.3.7 (This Specification number is not used.)

3/4.3.4 (This Specification number is not used.)

SURVEILLANCE REQUIREMENTS (Continued)

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

e. At least once per 18 months by:

1. Verifying that the pressure drop for the combined HEPA filters and charcoal adsorber banks is less than 5.6 inches Water Gauge while operating the pressurization filtration system at a flow rate of 800 to 1000 cfm.

2. Verifying that on a Containment Isolation Phase B/ Control Room High Radiation test signal, the system automatically closes all the series isolation ventilation system dampers which isolate the control room from the outside atmosphere and the system automatically starts 60 minutes later and supplies air to the control room through the HEPA filters and charcoal adsorber banks.

3. Deleted

4. Verifying that the pressurization filtration system maintains the control room at a positive pressure of ≥ 1/8 inch Water Gauge relative to the outside atmosphere during system operation.

5. Verifying that the heaters dissipate 5 ± 0.5 kw when tested in accordance with ANSI N510-1980.

6. Verifying that a control room high radiation/ containment phase B isolation signal will initiate operation of the bottled air pressurization system.

7. Verifying by a partial discharge test from four out of five sub-systems of the bottled air pressurization system at a discharge flow of less than 1000 cfm that the bottled air pressurization system will pressurize the control room to ≥ 1/8 inch Water Gauge relative to the outside atmosphere during system operation.

3/4.9.7 (This Specification number is not used.)

BASES

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Protection System and Engineered Safety Feature Actuation System Instrumentation and interlocks ensure that 1) the associated action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation, and 3) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses. The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

OPERABILITY of the following trips in Table 3.3-1 provides additional diverse or anticipatory protection features and is not credited in the accident analyses:

Undervoltage - Reactor Coolant Pumps (Above P-7); Underfrequency Reactor Coolant Pumps (Above P-7); Turbine Trip (Above P-9); Reactor Coolant Pump Breaker Position Trip (Above P-7); Turbine Impulse Chamber Pressure, P-13.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and supplements to that report as approved by the NRC and documented in the SER (letter to J. J. Sheppard from Cecil O. Thomas dated February 21, 1985). Jumpers and lifted leads are not an acceptable method for placing equipment in bypass as documented in the NRC safety evaluation report for this WCAP.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The Engineered Safety Feature Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A setpoint is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the setpoints have been specified in Table 3.3-4. Operation with setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its trip setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the trip setpoint and the value used in the analysis for the actuation. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified trip setpoint. S or Sensor Drift is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

shutdown because the unit must be in at least MODE 3 to perform the test. The P-6 permissive neutron detector CHANNEL CALIBRATION is performed in conjunction with the intermediate range neutron detectors. The overtemperature ΔT, P-8, P-9 and P-10 permissive neutron detector CHANNEL CALIBRATIONS are performed in conjunction with the power range neutron detectors.

Source Range Neutron Flux

The limiting condition for operation (LCO) requirement for the source range neutron flux trip function ensures that protection is provided against an uncontrolled rod cluster control assembly (RCCA) bank rod withdrawal accident from a subcritical condition during startup with the reactor trip breakers (RTBs) closed. This trip function provides redundant protection to the Power Range Neutron Flux-Low Setpoint and Intermediate Range Neutron Flux trip functions (see UFSAR Section 15.4.1 and Specification 2.2.1 Bases). In MODES 3, 4, and 5, with the RTBs closed, administrative controls also prevent the uncontrolled withdrawal of rods. The nuclear instrumentation system (NIS) source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. In Modes 3, 4, and 5, with the reactor trip breakers closed, the source range detectors provide an automatic trip function with a setpoint in the shutdown range and the intermediate range detectors provide an automatic trip function with a setpoint in the power range. Therefore, the functional capability at the specified trip setpoint is assumed to be available.

The LCO requires two channels of source range neutron flux to be OPERABLE when the RTBs are closed. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip function. The LCO also requires one channel of the source range neutron flux to be OPERABLE in MODE 3, 4, or 5 with RTBs open. In this case, the source range function is to provide control room indication and the high flux at shutdown alarm. The outputs of the function to RTS logic are not required OPERABLE when the RTBs are open.

The source range neutron flux function provides protection for control rod withdrawal from subcritical, boron dilution and control rod ejection events. The function also provides visual neutron flux indication in the control room.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

CHANNEL CALIBRATION

The alternate source range detectors are modified by a note to indicate they are not subject to the source range detector surveillance requirements until they have been connected to the applicable circuits and are required to be OPERABLE. This complies with the testing requirements for components that are required to be OPERABLE.

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. The CHANNEL CALIBRATION for the source range neutron detectors consists of obtaining the detector plateau and preamp discriminator curves, evaluating those curves, and establishing detector operating conditions as directed by the detector manufacturer. The 18 month frequency is based on the need to perform this surveillance under the conditions that apply during a plant outage since performance at power is not possible. The protection and monitoring functions are also calibrated at an 18 month frequency as is normal for reactor protection instrument channels. Operating experience has shown these components usually pass the surveillance when performed on the 18 month frequency.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: 1) the radiation levels are continually measured in the areas served by the individual channels; 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and 3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of NUREG-0737, "Clarification of TMI Action Plan Requirements," October, 1980.

3/4.3.3.2 (This Specification number is not used.)

3/4.3.3.3 (This Specification number is not used.)

3/4.3.3.4 (This Specification number is not used.)

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.6 (This Specification number is not used).

3/4.3.3.7 (This Specification number is not used).

3/4.3.3.8 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

3/4.3.3.11 EXPLOSIVE GAS MONITORING INSTRUMENTATION

This instrumentation includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.4 (This Specification number is not used.)

3/4.7.5 ULTIMATE HEAT SINK (Continued)

exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants."

3/4.7.6 FLOOD PROTECTION

The limitation on flood level ensures that facility operation will be terminated in the event of flood conditions. The limit of elevation 695 Mean Sea Level was selected on an arbitrary basis as an appropriate flood level at which to terminate further operation and initiate flood protection measures for safety related equipment.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP AND PRESSURIZATION SYSTEM

The OPERABILITY of the control room emergency air cleanup and pressurization system ensures that the control room will remain habitable with respect to potential radiation hazards for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

The control room air cleanup system includes two pressurization systems. The filtration pressurization system draws outside air through filters. The bottled air pressurization system pressurizes by discharge of air from bottles without filtration and with closure of intake and exhaust dampers. Although the bottles are shared with Unit 1, the discharge can be initiated by Unit 2 control systems in response to radiation levels. Closure of the intake and exhaust dampers can be initiated by Unit 2 control systems. However, closure of dampers in one intake and in one exhaust is dependent upon availability of Unit 1 power sources.

3/4.7.8 SUPPLEMENTAL LEAK COLLECTION AND RELEASE SYSTEM (SLCRS)

The OPERABILITY of the SLCRS provides for the filtering of postulated radioactive effluents resulting from a Fuel Handling Accident (FHA) and from leakage of loss of coolant accident (LOCA) activity from systems outside of the Reactor Containment building, such as Engineered Safeguards Features (ESF) equipment, prior to their release to the environment. This system also collects potential leakage of LOCA activity from the Reactor Containment building

3/4.7.8 SUPPLEMENTAL LEAK COLLECTION AND RELEASE SYSTEM (SLCRS) (Continued)

penetrations into the contiguous areas ventilated by the SLCRS except for the Emergency Air Lock. The operation of this system was assumed in calculating the postulated offsite doses in the analysis for a FHA. System operation was also assumed in that portion of the Design Basis Accident (DBA) LOCA analysis which addressed ESF leakage following the LOCA, however, no credit for SLCRS operation was taken in the DBA LOCA analysis for collection and filtration of Reactor Containment building leakage even though an unquantifiable amount of contiguous area penetration leakage would in fact be collected and filtered. Based on the results of the analyses, the SLCRS must be OPERABLE to ensure that ESF leakage following the postulated DBA LOCA and leakage resulting from a FHA will not exceed 10 CFR 100 limits.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on sealed source contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. Leakage of sources excluded from the requirements of this specification represent less than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested.

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.10 and 3/4.7.11 RESIDUAL HEAT REMOVAL SYSTEM (RHR)

Deleted

3/4.7.12 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other similar event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies based upon the number of unacceptable snubbers found during the previous inspection, the total population or category size for each type of snubber, and the previous inspection interval. This criteria follows the guidance provided in NRC Generic Letter 90-09. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, or verified OPERABLE by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at refueling or 18 month intervals not to exceed two (2) years. Observed failures of these sample snubbers shall require functional testing of additional units.

3/4.7.12 SNUBBERS (Continued)

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc...). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.13 STANDBY SERVICE WATER SYSTEM (SWE)

The OPERABILITY of the SWE ensures that sufficient cooling capacity is available to bring the reactor to a cold shutdown condition in the event that a barge explosion at the station's intake structure or any other extremely remote event would render all of the normal Service Water System (SWS) supply pumps inoperable. The scenario of a postulated gasoline barge impact with the intake structure and coincident explosion disabling the SWS is a low probability event. Nonetheless, the SWE provides defense in-depth in assuring shutdown cooling capability. The requirement to operate the SWE is not coincident with a postulated Design Basis Accident, but only for the postulated gasoline barge impact event.

Although the SWE is a non-safety system which is not required to meet single active failure criteria, the system is designed with redundant pumps and valves on a header to accommodate a single active failure on start-up. This design criteria provides a defense in-depth in order to ensure the system can adequately mitigate the consequences of the postulated event. An SWE pump can be manually started on the emergency bus during loss of offsite power after the diesel loading sequence is complete. With no loss of power signal present, the SWE is automatically started upon receipt of low service water header pressure signal. This feature is provided to prevent inadvertent plant trip on loss of running service water pump and is not required

3/4.7.13 STANDBY SERVICE WATER SYSTEM (SWE) (Continued)

for the design basis event. If there is a delay in starting the SWE, the auxiliary feedwater system is available to remove reactor core decay heat for a short term period.

The requirements for subsystem OPERABILITY are similar to those of the SWS except that one subsystem is required to be OPERABLE in the MODES noted. The Limiting Condition for Operation reflects the low risk of the postulated event compared to more stringent requirements associated with safety related systems. The ACTION statement takes into account the low probability of both trains of SWS being disabled as a result of the postulated scenario coincident with one of the SWE subsystems being OPERABLE.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS (Continued)

requirements are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

All containment penetrations, except for the containment purge and exhaust penetrations, that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Penetration closure may be achieved by an isolation valve, blind flange, manual valve, or functional equivalent. Functional equivalent isolation ensures releases from the containment are prevented for credible accident scenarios. The isolation techniques must be approved by an engineering evaluation and may include use of a material that can provide a temporary, pressure tight seal capable of maintaining the integrity of the penetration to restrict the release of radioactive material from a fuel element rupture.

3/4.9.5 COMMUNICATIONS

The requirements for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE OPERABILITY

The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies; 2) each crane has sufficient load capacity to lift a control rod or fuel assembly; and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 (This Specification is not used.)

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor