ARKANSAS NUCLEAR ONE - UNIT 1

IMPROVED
TECHNICAL SPECIFICATIONS
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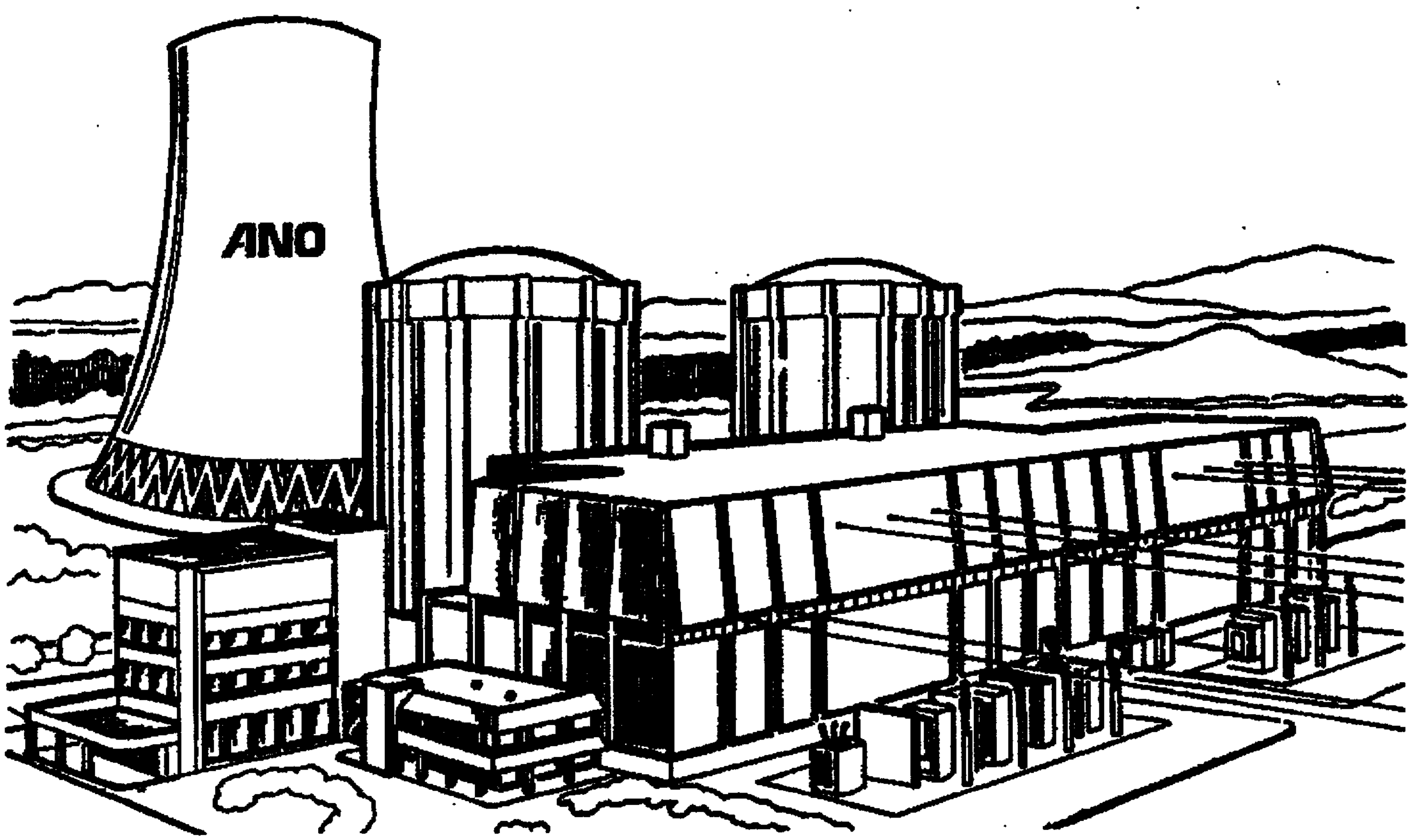

05/01/01 Supplement
Volume 1 of 2
(Sections 3.3A, 3.3B, 3.3C, and 3.3D)

Entergy

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3.3 INSTRUMENTATION

3.3.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1 Four channels of RPS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Place channel in bypass or trip. <u>OR</u> A.2 Prevent bypass of remaining channels.	1 hour 1 hour
B. Two channels inoperable.	B.1 Place one channel in trip. <u>AND</u> B.2.1 Place second channel in bypass. <u>OR</u> B.2.2 Prevent bypass of remaining channels.	1 hour 1 hour 1 hour
C. Three or more channels inoperable. <u>OR</u> Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the Condition referenced in Table 3.3.1-1 for the Function.	Immediately

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. As required by Required Action C.1 and referenced in Table 3.3.1-1.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Open all control rod drive (CRD) trip breakers.	6 hours
E. As required by Required Action C.1 and referenced in Table 3.3.1-1.	E.1	Open all CRD trip breakers.	6 hours
F. As required by Required Action C.1 and referenced in Table 3.3.1-1.	F.1	Reduce THERMAL POWER < 45% RTP.	6 hours
G. As required by Required Action C.1 and referenced in Table 3.3.1-1.	G.1	Reduce THERMAL POWER < 10% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.1-1 to determine which SRs apply to each RPS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	-----NOTES----- 1. Adjust power range channel output if the absolute difference is > 2% RTP. 2. Not required to be performed until 24 hours after THERMAL POWER is ≥ 20% RTP. ---------- Compare results of calorimetric heat balance calculation to power range channel output.	96 hours <u>AND</u> Once within 24 hours after a THERMAL POWER change of ≥ 10% RTP
SR 3.3.1.3	-----NOTES----- 1. Adjust the power range channel imbalance output if the absolute value of the imbalance error is ≥ 2% RTP. 2. Not required to be performed until 24 hours after THERMAL POWER is ≥ 20% RTP. ---------- Compare results of out of core measured AXIAL POWER IMBALANCE to incore measured AXIAL POWER IMBALANCE.	31 days
SR 3.3.1.4	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.1.5	-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION.	18 months

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Nuclear Overpower -				
a. High Setpoint	1,2[(a)],3[(d)]	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.4 SR 3.3.1.5	≤ 104.9% RTP
b. Low Setpoint	2[(b)],3[(b)] 4[(b)],5[(b)]	E	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 5% RTP
2. RCS High Outlet Temperature	1,2	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 618°F
3. RCS High Pressure	1,2[(a)],3[(d)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 2355 psig
4. RCS Low Pressure	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≥ 1800 psig
5. RCS Variable Low Pressure	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	As specified in the COLR
6. Reactor Building High Pressure	1,2,3[(c)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 18.7 psia
7. Reactor Coolant Pump to Power	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 55% RTP with one pump operating in each loop.
8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5	As specified in the COLR
9. Main Turbine Trip (Oil Pressure)	≥ 45% RTP	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≥ 40.5 psig
10. Loss of Main Feedwater Pumps (Control Oil Pressure)	≥ 10% RTP	G	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≥ 55.5 psig
11. Shutdown Bypass RCS High Pressure	2[(b)],3[(b)] 4[(b)],5[(b)]	E	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5	≤ 1720 psig

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) With any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

3.3 INSTRUMENTATION

3.3.2 Reactor Protection System (RPS) Manual Reactor Trip

LCO 3.3.2 The RPS Manual Reactor Trip Function shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with any control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Manual Reactor Trip Function inoperable.	A.1 Restore Function to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. <u>AND</u> B.2 Open all CRD trip breakers.	6 hours 6 hours
C. Required Action and associated Completion Time not met in MODE 4 or 5.	C.1 Open all CRD trip breakers.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL FUNCTIONAL TEST.	Once prior to each reactor startup if not performed within the previous 7 days

3.3 INSTRUMENTATION

3.3.3 Reactor Protection System (RPS) - Reactor Trip Module (RTM)

LCO 3.3.3 Four RTMs shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with any control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RTM inoperable.	A.1.1 Open the associated CRD trip breaker.	1 hour
	<u>OR</u>	
	A.1.2 Remove power from the associated CRD trip breaker.	1 hour
	<u>AND</u>	
	A.2 Physically remove the inoperable RTM.	1 hour
B. Two or more RTMs inoperable in MODE 1, 2, or 3. <u>OR</u> Required Action and associated Completion Time not met in MODE 1, 2, or 3.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2.1 Open all CRD trip breakers.	6 hours
	<u>OR</u>	
	B.2.2 Remove power from all CRD trip breakers.	6 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two or more RTMs inoperable in MODE 4 or 5. OR Required Action and associated Completion Time not met in MODE 4 or 5.	C.1 Open all CRD trip breakers. OR C.2 Remove power from all CRD trip breakers.	6 hours 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform CHANNEL FUNCTIONAL TEST.	92 days

3.3 INSTRUMENTATION

3.3.4 Control Rod Drive (CRD) Trip Devices

LCO 3.3.4 The following CRD trip devices shall be OPERABLE:

a. Two AC CRD trip breakers;

b. Two DC CRD trip breaker pairs; and

c. Eight electronic trip assembly (ETA) relays.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each CRD trip device.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more CRD trip breaker(s) or breaker pair undervoltage or shunt trip Functions inoperable.	A.1 Open the CRD trip breaker. OR A.2 Remove power from the CRD trip breaker.	48 hours 48 hours
B. One or more CRD trip breaker(s) or breaker pair inoperable for reasons other than those in Condition A.	B.1 Open the CRD trip breaker. OR B.2 Remove power from the CRD trip breaker.	1 hour 1 hour

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more required ETA relays inoperable.	C.1	Transfer affected CONTROL ROD group to power supply with OPERABLE or open ETA relays.	1 hour
	OR		
	C.2	Transfer affected CONTROL ROD group to a DC hold power supply.	1 hour
	OR		
	C.3	Place the SCRs associated with the inoperable ETA relay in trip.	1 hour
	OR		
	C.4	Open corresponding AC CRD trip breaker.	1 hour
D. Required Action and associated Completion Time not met in MODE 1, 2, or 3.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2.1	Open all CRD trip breakers.	6 hours
	OR		
	D.2.2	Remove power from all CRD trip breakers.	6 hours
E. Required Action and associated Completion Time not met in MODE 4 or 5.	E.1	Open all CRD trip breakers.	6 hours
	OR		
	E.2	Remove power from all CRD trip breakers.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform CHANNEL FUNCTIONAL TEST.	92 days

3.3 INSTRUMENTATION

3.3.9 Source Range Neutron Flux

LCO 3.3.9 One source range neutron flux channel shall be OPERABLE.

APPLICABILITY: MODES 2, 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required source range neutron flux channel inoperable with ≤ 1E-10 amp on the intermediate range neutron flux channel.	---------------------NOTE------------------ Plant temperature changes are allowed provided the temperature change is accounted for in the SDM calculations. ---	
	A.1 Suspend operations involving positive reactivity changes.	Immediately
	AND	
	A.2 Initiate action to insert all CONTROL RODS.	Immediately
	AND	
	A.3 Open control rod drive trip breakers.	1 hour
	AND	
	A.4 Verify SDM to be within the limit provided in the COLR.	1 hour AND Once per 12 hours thereafter
B. Required source range neutron flux channel inoperable with > 1E-10 amp on the intermediate range neutron flux channel.	B.1 Initiate action to restore required channel to OPERABLE status.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.9.2	---NOTE--- Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	18 months

3.3 INSTRUMENTATION

3.3.10 Intermediate Range Neutron Flux

LCO 3.3.10 One intermediate range neutron flux channel shall be OPERABLE.

APPLICABILITY: MODE 2,
MODES 3, 4, and 5 with any control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required channel inoperable.	---------------------NOTE------------------ Plant temperature changes are allowed provided the temperature change is accounted for in the SDM calculations. ---	
	A.1 Suspend operations involving positive reactivity changes.	Immediately
	AND	
	A.2 Open CRD trip breakers.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.10.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.10.2	Perform CHANNEL FUNCTIONAL TEST.	31 days

SURVEILLANCE		FREQUENCY
SR 3.3.10.3	------------------------------NOTE------------------------------ Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	 18 months

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor trip, if necessary, to protect against violating the core fuel design limits and the Reactor Coolant System (RCS) pressure boundary during abnormalities. By tripping the reactor, the RPS also assists the Engineered Safety Feature (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by identifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as the LCOs and administrative controls on other parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs and administrative controls, establishes the threshold for protective system action to prevent exceeding specified acceptable limits during Design Basis Accidents (DBAs). Acceptable consequences for accidents are that the offsite dose shall be maintained within 10 CFR 100 limits or other limits approved by the NRC. During abnormalities, one or more of the following limits is maintained:

a. For accidents other than locked rotor, the departure from nucleate boiling ratio (DNBR) shall be maintained above the Safety Limit (SL) value. For the locked rotor accident, the minimum DNBR shall not be less than the applicable critical heat flux correlation limit, or fuel cladding shall be shown to experience no significant temperature excursions;

b. Fuel centerline temperature shall be maintained below the SL value;

c. The RCS pressure SL of 2750 psig shall not be exceeded; and

d. Reactor power shall not exceed 112% RTP.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 100 criteria during abnormalities.

RPS Overview

The RPS consists of four separate redundant protection channels that receive inputs of neutron flux, RCS pressure, RCS flow, reactor outlet temperature, RCS pump status, reactor building (RB) pressure, main feedwater (MFW) pump turbine status, and main turbine status.

Figure 7.1, SAR, Chapter 7 (Ref. 1), shows the arrangement of the RPS protection channels. A protection channel is composed of measurement channels, a manual trip channel, a reactor trip module (RTM), and control rod drive (CRD) trip devices. LCO 3.3.1 provides requirements for the individual measurement channels. These channels encompass all equipment and electronics from the point at which the measured parameter is sensed through the bistable relay contacts in the trip string. LCO 3.3.2, "Reactor Protection System (RPS) Manual Reactor Trip," LCO 3.3.3, "Reactor Protection System (RPS) - Reactor Trip Module (RTM)," and LCO 3.3.4, "Control Rod Drive (CRD) Trip Devices," discuss the remaining RPS elements.

The RPS instrumentation measures critical unit parameters and compares these to predetermined setpoints. If the setpoint is exceeded, a channel trip signal is generated. The generation of trip signals in any two of the four RPS channels will result in the trip of the reactor.

The Reactor Trip System (RTS) contains multiple CRD trip devices, two AC trip breakers, and two DC trip breaker pairs that provide a path for power to the CRD System. In addition to the safety rods, the power for the regulating rods and APSRs may be interrupted by the electronic trip assembly (ETA) relays. The system has two separate paths (or channels), with each path having either two breakers in series or a breaker and an ETA relay controlled silicon controlled rectifier (SCR) in series. Each path provides independent power to the CRDs. Either path can provide sufficient power to operate all CRDs. Two separate power paths to the CRDs ensure that a single failure that opens one path will not cause an unwanted reactor trip.

The RPS consists of four independent protection channels, each containing an RTM. Each RTM receives signals from its own measurement channels that indicate a protection channel trip is required. The RTM transmits this signal to its own two-out-of-four trip logic and to the two-out-of-four logic of the RTMs in the other three RPS channels. Whenever any two RPS channels trip, the RTM in each channel actuates to remove 120 VAC power from its associated CRD trip breaker.

The reactor is tripped by opening circuit breakers and de-energizing ETA relays that interrupt the control power supply to the CRDs. Six breakers are installed to increase reliability and allow testing of the trip system. A one-out-of-two taken twice logic is used to interrupt power to the rods.

The RPS has two manual bypasses: a shutdown bypass and a channel bypass. Shutdown bypass allows the withdrawal of safety rods to provide the availability of rapidly insertable negative reactivity during unit cooldowns or heatups. Channel bypass is typically used for maintenance and testing. Test circuits in the trip strings allow testing of RPS trip Functions. Also, an automatic bypass is provided at low power levels for the Main Turbine Trip and the Loss of Main Feedwater Pump Functions.

The RPS receives input from the instrumentation channels discussed next. The specific relationship between measurement channels and protection channels differs from parameter to parameter.

These arrangements and the relationship of instrumentation channels to trip Functions are discussed below to assist in understanding the overall effect of instrumentation channel failure.

Power Range Nuclear Instrumentation

Power Range Nuclear Instrumentation channels provide inputs to the following RPS trip Functions:

1. Nuclear Overpower
    a. Nuclear Overpower - High Setpoint;
    b. Nuclear Overpower - Low Setpoint;

7. Reactor Coolant Pump to Power;

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE (Power Imbalance Flow);

9. Main Turbine Trip (Oil Pressure); and

10. Loss of Main Feedwater Pumps (Control Oil Pressure).

The Main Turbine Trip and Loss of Main Feedwater Pumps Functions utilize the Power Range Nuclear Instrumentation only for enabling/disabling the operating bypass at low power levels.

The power range instrumentation has four linear channels, one for each core quadrant. Each channel feeds one RPS protection channel. Each channel originates in a detector assembly containing two uncompensated ion chambers. The ion chambers are positioned to represent the top half and bottom half of the core. The individual currents from the chambers are fed to individual linear amplifiers. The summation of the top and bottom is the total reactor power. The top minus the bottom neutron signal is the measured AXIAL POWER IMBALANCE of the reactor core.

Reactor Outlet Temperature

The Reactor Outlet Temperature provides input to the following Functions:

2. Reactor Outlet High Temperature; and

5. RCS Variable Low Pressure.

The Reactor Outlet Temperature is measured by two resistance elements in each hot leg, for a total of four. One temperature detector is associated with each protection channel.

Reactor Coolant System Pressure

The Reactor Coolant System Pressure provides input to the following Functions:

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure; and

11. Shutdown Bypass RCS High Pressure.

The RPS inputs of reactor coolant pressure are provided by two pressure transmitters in each hot leg, for a total of four. One sensor is associated with each protection channel.

Reactor Building Pressure

The Reactor Building Pressure measurements provide input only to the Reactor Building High Pressure trip, Function 6. There are four RB High Pressure sensors, one associated with each protection channel.

Reactor Coolant Pump Power Monitoring

Reactor coolant pump power monitors are inputs to the Reactor Coolant Pump to Power trip, Function 7. Each RCP's operating current is measured by a current transformer providing the current input to the associated RCP underpower relay, and the bus voltage is measured by a potential transformer providing the voltage input to the associated RCP underpower relays. Each RCP underpower relay provides individual RCP status to each protection channel.

Reactor Coolant System Flow

The Reactor Coolant System Flow measurements are an input to the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip, Function 8. The reactor coolant flow inputs to the RPS are provided by eight differential pressure transmitters, four on each loop, which measure flow through calibrated flow tubes. One flow input in each loop is associated with each protection channel.

Main Turbine Automatic Stop Oil Pressure

Main Turbine Automatic Stop Oil Pressure is an input to the Main Turbine Trip (Oil Pressure) reactor trip, Function 9. Each of the four protection channels receives turbine status information from one of four pressure switches monitoring main turbine automatic stop oil pressure. Contact buffers in each protection channel continuously monitor the status of the contact inputs and initiate an RPS trip when a main turbine trip is indicated.

Feedwater Pump Control Oil Pressure

Feedwater Pump Control Oil Pressure is an input to the Loss of Main Feedwater Pumps (Control Oil Pressure) trip, Function 10. Control oil pressure is measured by four switches on each feedwater pump. One switch on each pump is associated with each protection channel.

RPS Bypasses

The RPS is designed with two types of manual bypasses: channel bypass and shutdown bypass.

Channel bypass provides a method of placing all Functions in one RPS protection channel in a bypassed condition, and shutdown bypass provides a method of leaving the safety rods withdrawn during cooldown and depressurization of the RCS. Each bypass is discussed next.

Channel Bypass

A channel bypass provision is provided to allow for maintenance and testing of the RPS. The use of channel bypass keeps the protection channel trip relay energized regardless of the status of the instrumentation channel bistable relay contacts. To place a protection channel in channel bypass, the key switch must be operated, and the other three channels must not be in channel bypass. This is ensured by contacts from the other channels being in series with the channel bypass relay. If any contact is open, the second channel cannot be bypassed. When the bypass relay is energized, the bypass contact closes, maintaining the channel trip relay in an energized condition. An indicator light remains lit while the channel is in bypass. All RPS trips are reduced to a two-out-of-three logic in channel bypass. Only one channel bypass key is accessible for use in the control room.

Shutdown Bypass

During unit cooldown, it is allowable to leave some safety rods withdrawn to provide shutdown capabilities in the event of unusual positive reactivity additions (moderator dilution, etc.).

However, the unit is also depressurized as coolant temperature is decreased. If the safety rods are withdrawn and coolant pressure is decreased, an RCS Low Pressure trip will occur at 1800 psig and the rods will fall into the core. To avoid this, the protection system allows the operator to bypass the low pressure trip and maintain shutdown capabilities. During the cooldown and depressurization, the safety rods are inserted prior to the low pressure trip of 1800 psig. The RCS pressure is decreased to less than 1720 psig, then each RPS channel is placed in shutdown bypass.

When an RPS channel is placed in shutdown bypass, the RCS Low Pressure trip, Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip, Reactor Coolant Pump to Power trip, and the RCS Variable Low Pressure trip, are bypassed and a RCS High Pressure, ≤ 1720 psig trip and a Nuclear Overpower

Low Setpoint trip, ≤ 5% RTP, are inserted. The operator can now withdraw the safety rods for additional rapidly insertable negative reactivity.

The insertion of the high pressure trip with a trip setpoint of ≤ 1720 psig prevents operation at normal system pressure, approximately 2155 psig, with a portion of the RPS bypassed, and ensures that the bypass is removed prior to normal operation. When the RCS pressure is increased during a unit heatup, the safety rods are inserted prior to reaching 1720 psig. The shutdown bypass is removed, which returns the RPS to normal, and system pressure is increased to greater than 1800 psig. All or some of the safety rods are then withdrawn and normally remain at the full out condition for the rest of the heatup.

The insertion of the Nuclear Overpower Low Setpoint Trip provides a backup to the Shutdown Bypass RCS High Pressure trip while preventing the generation of any significant amount of power.

Module Interlock and Test Trip Relay

Each channel and each trip module is capable of being individually tested. When a module is placed into the test mode, it causes the test trip relay to open and to indicate an RPS channel trip. Under normal conditions, the channel to be tested is placed in bypass before a module is tested.

Trip Setpoints/Allowable Value

The trip setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy.

The trip setpoints used in the bistables are based on the analytical limits used in the safety analysis described in SAR, Chapter 14 and Chapter 3A (Ref. 2). The selection of these trip setpoints is such that adequate protection is provided when appropriate sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and environment errors, the Allowable Values specified in Table 3.3.1-1 are equal to or conservatively adjusted with respect to the analytical limits. Guidance used to calculate the uncertainty associated with the trip setpoints is provided in Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001 (Ref. 3). The explicit uncertainties are addressed in the individual design calculations as required. The trip setpoint entered into the bistable may be more conservative than that specified by the Allowable Value to account for changes in instrument error detectable by a CHANNEL FUNCTIONAL TEST. A channel is inoperable if its as-found trip setpoint is not within its required Allowable Value.

Setpoints in accordance with the Allowable Value ensure that the limits of Chapter 2.0, "Safety Limits," in the Technical Specifications are not violated during abnormalities and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the abnormality or DBA and the equipment functions as analyzed. Note that in LCO 3.3.1 the Allowable Values listed in Table 3.3.1-1 are the LSSS.

Each channel can be tested online to verify that the signal and trip setpoint are within the specified allowance requirements of approved calibration procedures. Once a designated channel is taken out of service for testing, a simulated signal may be injected in place of the field instrument signal. The process equipment for the channel may then be tested, verified, and calibrated.

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

Analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in the SAR, Chapter 14 and Chapter 3A (Ref. 2), takes credit for most RPS trip Functions. Functions not specifically credited in the accident analysis were qualitatively credited in the NRC staff approved licensing basis for the unit. These Functions are high RB pressure, high RCS temperature, turbine trip, loss of main feedwater, the shutdown bypass nuclear overpower low setpoint, and shutdown bypass high pressure. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions also serve as backups to Functions that were credited in the safety analysis.

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable. The four channels of each Function in Table 3.3.1-1 of the RPS instrumentation shall be OPERABLE during its specified Applicability to ensure that a reactor trip will be actuated if needed. Additionally, during shutdown bypass with any CRD trip breaker closed, the applicable RPS Functions must also be OPERABLE. This ensures the capability to trip the withdrawn CONTROL RODS exists at all times that rod motion is possible. The trip Function channels specified in Table 3.3.1-1 are considered OPERABLE when all channel components necessary to provide a reactor trip are functional and in service for the required MODE or Other Specified Condition listed in Table 3.3.1-1.

Required Actions allow maintenance (protection channel) bypass of individual channels, but the bypass activates interlocks that prevent operation with a second channel bypass. Bypass effectively places the unit in a two-out-of-three logic configuration that can still initiate a reactor trip, even with a single failure within the system.

Only the Allowable Values are specified for each RPS trip Function in the LCO. Trip setpoints are specified in the setpoint calculations or calibration procedures. The setpoints are selected such that the setpoint measured by CHANNEL FUNCTIONAL TESTS is not expected to exceed the Allowable Value if the bistable is performing as required.

For most RPS Functions, the Allowable Value is to ensure that the departure from nucleate boiling (DNB) or RCS pressure SLs are not challenged. Cycle specific figures for use during operation are contained in the COLR.

Certain RPS trips function to indirectly protect the SLs by detecting specific conditions that do not immediately challenge SLs but will eventually lead to challenge if no action is taken. These trips function to minimize the consequences of unit transients caused by the specific conditions. The Allowable Value for these Functions is selected at the specified deviation from normal values that will indicate the condition, without risking spurious trips due to normal fluctuations in the measured parameter.

The Allowable Values for bypass removal Functions are stated in the Applicable MODE or Other Specified Condition column of Table 3.3.1-1.

The safety analyses applicable to each RPS Function are discussed next.

1. Nuclear Overpower

   a. Nuclear Overpower - High Setpoint

      The Nuclear Overpower - High Setpoint trip provides protection for the design thermal overpower condition based on the measured out of core fast neutron leakage flux.

      The Nuclear Overpower - High Setpoint trip initiates a reactor trip when the neutron power reaches a predefined setpoint at the design overpower limit. Because THERMAL POWER lags the neutron power, tripping when the neutron power reaches the design overpower will limit THERMAL POWER to a maximum value of the design overpower.

      Thus, the Nuclear Overpower - High Setpoint trip protects against violation of the DNBR and fuel centerline melt SLs. However, the RCS Variable Low Pressure, and Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE, also provide protection. The role of the Nuclear Overpower - High Setpoint trip is to limit reactor THERMAL POWER below the highest power at which the other two trips are known to provide protection.

      The Nuclear Overpower - High Setpoint trip also provides transient protection for rapid positive reactivity excursions during power operations. These events include the rod withdrawal accident, the rod ejection accident, and the steam line break accident. By providing a trip during these events, the Nuclear Overpower - High Setpoint trip protects the unit from excessive power levels and also serves to reduce reactor power to prevent violation of the RCS pressure SL.

      Rod withdrawal accident analyses cover a large spectrum of reactivity insertion rates (rod worths), which exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower - High Setpoint trip provides the primary protection. At low reactivity insertion rates, the high pressure trip provides primary protection.

The specified Allowable Value is selected to initiate a trip at or before reactor power exceeds the highest point at which the RCS Variable Low Pressure and the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trips are analyzed to provide protection against DNB and fuel centerline melt. The Allowable Value does not account for harsh environment induced errors, because the trip will actuate prior to degraded environmental conditions being reached.

b. Nuclear Overpower - Low Setpoint

While in shutdown bypass, the Nuclear Overpower - Low Setpoint is instated with a trip setpoint of ≤ 5% RTP. The low power setpoint, in conjunction with the Shutdown Bypass RCS High Pressure setpoint, protect the unit from excessive power conditions when other RPS trips are bypassed.

The Allowable Value was chosen to be as low as practical and still lie within the range of the out of core instrumentation.

2. Reactor Outlet High Temperature

The Reactor Outlet High Temperature trip, in conjunction with the RCS Low Pressure and RCS Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the reactor outlet temperature approaches the conditions necessary for DNB. Portions of each Reactor Outlet High Temperature trip channel are common with the RCS Variable Low Pressure trip. The Reactor Outlet High Temperature trip provides steady state protection for the DNBR SL.

The Reactor Outlet High Temperature trip limits the maximum RCS temperature to below the highest value for which DNB protection by the Variable Low Pressure trip is ensured. The trip setpoint Allowable Value is selected to initiate a trip before hot leg temperatures reach the point beyond which the RCS Low Pressure and Variable Low Pressure trips are analyzed. Above the high temperature trip, the variable low pressure trip need not provide protection, because the unit would have tripped already. The Allowable Value does not reflect errors induced by harsh environmental conditions that the equipment is expected to experience because the trip will actuate prior to degraded environmental conditions being reached.

3. RCS High Pressure

The RCS High Pressure trip works in conjunction with the pressurizer safety valves to prevent RCS overpressurization, thereby protecting the RCS High Pressure SL.

The RCS High Pressure trip has been credited in the accident analysis calculations for slow positive reactivity insertion transients (rod withdrawal accidents and moderator dilution). The rod withdrawal accidents cover a large

spectrum of reactivity insertion rates and rod worths that exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower - High Setpoint trip provides the primary protection. At low reactivity insertion rates, the RCS High Pressure trip provides the primary protection.

The Allowable Value is selected such that the RCS High Pressure SL is not exceeded during steady state operation or slow power increasing transients. The Allowable Value does not reflect errors induced by harsh environmental conditions because the trip will actuate prior to degraded environmental conditions being reached.

4. RCS Low Pressure

The RCS Low Pressure trip, in conjunction with the Reactor Outlet High Temperature and Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated prior to reactor outlet temperature exceeding the conditions necessary for DNB. The RCS Low Pressure trip provides the DNB low pressure limit for the RCS Variable Low Pressure trip.

The RCS Low Pressure Allowable Value is selected to initiate a reactor trip before RCS pressure is reduced below the lowest point at which the RCS Variable Low Pressure trip is analyzed. The RCS Low Pressure trip provides protection for primary system depressurization events and has been credited in the accident analysis calculations for small break loss of coolant accidents (LOCAs). Consequently, harsh RB conditions created by small break LOCAs can affect performance of the RCS pressure sensors and transmitters. Therefore, degraded environmental conditions are considered in the Allowable Value determination.

5. RCS Variable Low Pressure

The RCS Variable Low Pressure trip, in conjunction with the Reactor Outlet High Temperature and RCS Low Pressure trips, provides protection for the DNBR SL. A trip is initiated prior to the system parameters of pressure and temperature exceeding the conditions necessary for DNB. The RCS Variable Low Pressure trip provides a floating low pressure trip based on the reactor outlet temperature expressed in degrees Fahrenheit within the range specified by the Reactor Outlet High Temperature and RCS Low Pressure trips.

The RCS Variable Low Pressure Allowable Value is selected to initiate a trip prior to temperature and pressure exceeding the conditions necessary for DNB while operating in a temperature pressure region constrained by the low pressure and high temperature trips. The RCS Variable Low Pressure trip is not assumed for transient protection in the unit safety analysis. Therefore, the Allowable Value does not account for errors induced by a harsh RB environment.

6. Reactor Building High Pressure

The Reactor Building High Pressure trip provides an early indication of a high energy line break (HELB) inside the RB. By detecting changes in the RB pressure, the RPS can provide a reactor trip before the other system parameters have varied significantly. Thus, this trip acts to minimize accident consequences.

The Allowable Value for RB High Pressure trip is set at the lowest value consistent with avoiding spurious trips during normal operation. Even in the case where this trip is a backup for other RPS trips for LOCA or MSLB, it is assumed to occur before degraded building conditions have an appreciable effect on RB High Pressure trip components. Therefore, determination of the Allowable Value does not account for errors induced by a harsh environment.

7. <u>Reactor Coolant Pump to Power</u>

The Reactor Coolant Pump to Power trip provides protection for changes in the reactor coolant flow due to the loss of multiple RCPs. Because the flow reduction lags loss of power indications due to the inertia of the RCPs, the trip initiates protective action earlier than a trip based on a measured flow signal.

The trip also prevents operation with both pumps in either coolant loop tripped. Under these conditions, core flow and core fluid mixing may be insufficient for adequate heat transfer. Thus, the Reactor Coolant Pump to Power trip functions to protect the DNBR and fuel centerline temperature SLs.

The Reactor Coolant Pump to Power trip has been credited in the accident analysis calculations for the loss of four RCPs. The trip also provides protection for the loss of a pump or pumps which would result in both pumps in a single steam generator loop being tripped.

The Allowable Value for the Reactor Coolant Pump to Power trip setpoint is selected to prevent normal power operation unless at least one RCP is operating in each loop. RCP status is monitored by power transducers associated with each pump. These relays indicate a loss of an RCP on underpower. The underpower Allowable Value is selected to reliably trip on loss of voltage to the RCPs. Neither the reactor power nor the pump power Allowable Value account for instrumentation errors caused by harsh environments because the trip Function is not required to respond to events that could create harsh environments around the equipment.

8. <u>Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE</u>

The Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip provides steady state protection for the reactor core SLs. A reactor trip is initiated prior to the core power, AXIAL POWER IMBALANCE, and reactor coolant flow conditions exceeding the DNB or fuel centerline temperature limits.

This trip supplements the protection provided by the Reactor Coolant Pump to Power trip, through the power to flow ratio, for loss of reactor coolant flow events. The power to flow ratio provides direct protection for the limiting loss of flow transient which is the loss of two RCPs from four pump operation. The imbalance portion of the trip is credited for steady state protection only.

The power to flow ratio of the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip also provides steady state protection to prevent reactor power from exceeding the allowable power when the primary system is operating with two or three pump flow. Thus, the power to flow ratio prevents overpower conditions similar to the Nuclear Overpower trip. This protection ensures that during reduced flow conditions the core power is maintained below that required to begin DNB.

The Allowable Value is selected to ensure that a trip occurs prior to core power, axial power peaking, and reactor coolant flow conditions reaching DNB or fuel centerline temperature limits. The Allowable Value for this Function is given in the unit COLR because the cycle specific core peaking changes affect the Allowable Value.

9. Main Turbine Trip (Oil Pressure)

The Main Turbine Trip Function trips the reactor when the main turbine is tripped at high power levels. The Main Turbine Trip Function provides an early reactor trip in anticipation of the loss of heat sink associated with a turbine trip. The Main Turbine Trip Function was added to the B&W designed units in accordance with NUREG-0737 (Ref. 4) following the Three Mile Island Unit 2 accident. The trip lowers the probability of an RCS electromatic relief valve (ERV) actuation for turbine trip cases.

Each of the four turbine oil pressure switches feeds one of the four protection channels through a buffer that continuously monitors the status of the contacts. Therefore, failure of any pressure switch affects only one protection channel.

For the Main Turbine Trip (Oil Pressure) bistable, the Allowable Value of ≥ 40.5 psig is selected to provide a trip whenever main turbine oil pressure drops below the normal operating range. The reactor power bypass is designed to automatically remove the turbine oil pressure trip function from the bypassed condition at < 45% RTP. Alarms are available to alert operators when the bypass function is enabled. Should the automatic bypass removal function fail such that the channel remains in the bypassed state, the channel must be considered inoperable at power levels of ≥ 45% RTP and the appropriate condition is entered. Failure of the automatic bypass removal feature alone or the inability to place the channel in a bypassed state when < 45% RTP does not constitute channel inoperability. The automatic bypass removal feature is tested to ensure its continued availability during the monthly CHANNEL FUNCTIONAL TEST. The turbine trip is not required to protect against events that can create a harsh environment in the turbine building.

Therefore, errors induced by harsh environments are not included in the determination of the setpoint Allowable Value.

10. Loss of Main Feedwater Pumps (Control Oil Pressure)

The Loss of Main Feedwater Pumps (Control Oil Pressure) trip provides a reactor trip at high power levels when both MFW pumps are tripped. The trip provides an early reactor trip in anticipation of the loss of heat sink associated with a loss of main feedwater. This trip was added in accordance with NUREG-0737 (Ref. 4) following the Three Mile Island Unit 2 accident. This trip provides a reactor trip for a loss of main feedwater to minimize challenges to the ERV.

For the feedwater pump control oil pressure bistable, the Allowable Value of ≥ 55.5 psig is selected to provide a trip whenever feedwater pump control oil pressure drops below the normal operating range. The reactor power bypass is designed to automatically remove the main feedwater pump oil pressure trip function from the bypassed condition at < 10% RTP. Alarms are available to alert operators when the bypass function is enabled. Should the automatic bypass removal function fail such that the channel remains in the bypassed state, the channel must be considered inoperable at power levels of ≥ 10% RTP and the appropriate condition is entered. Failure of the automatic bypass removal feature alone or the inability to place the channel in a bypassed state when < 10% RTP does not constitute channel inoperability. The automatic bypass removal feature is tested to ensure its continued availability during the monthly CHANNEL FUNCTIONAL TEST. The Loss of Main Feedwater Pumps (Control Oil Pressure) trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors caused by harsh environments are not included in the determination of the setpoint Allowable Value.

11. Shutdown Bypass RCS High Pressure

The RPS Shutdown Bypass is provided to allow for withdrawing the CONTROL RODS while operating below the normal RCS Low Pressure trip setpoint. The shutdown bypass allows the operator to withdraw the safety groups of CONTROL RODS. This makes their negative reactivity available to terminate inadvertent reactivity excursions. Because the shutdown bypass high pressure trip setpoint is below the normal RCS low pressure trip setpoint, the reactor must be tripped while passing between these two setpoints. This ensures that RPS trips cannot be bypassed unless the CONTROL RODS are all inserted.

Accidents analyzed in the SAR, Chapter 14 and Chapter 3A (Ref. 2), do not include events that occur during shutdown bypass operation.

During shutdown bypass operation with the Shutdown Bypass RCS High Pressure trip active with a setpoint of ≤ 1720 psig and the Nuclear Overpower - Low Setpoint active with a setpoint of ≤ 5% RTP, the trips listed below are bypassed.

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE.

The Shutdown Bypass Nuclear Overpower - Low Setpoint Allowable Value is selected to initiate a trip before producing significant THERMAL POWER.

General Discussion

In MODES 1 and 2, the RPS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 5). In MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal, the RPS satisfies Criterion 4 of 10 CFR 50.36.

In MODE 1; in MODE 2, when not operating in shutdown bypass; and in MODE 3, when not operating in shutdown bypass but with any CRD trip breaker in the closed position and the CRD system capable of rod withdrawal, the following trips are required to be OPERABLE. These trips function to ensure that any withdrawn CONTROL RODS can be automatically inserted to make or maintain the reactor subcritical.

1.a. Nuclear Overpower-High Setpoint; and

3. RCS High Pressure.

In MODES 1 and 2, the following trips are required to the OPERABLE. These trips function as primary or as back-up trips to ensure that any withdrawn CONTROL RODS can be automatically inserted to make or maintain the reactor subcritical.

2. Reactor Outlet High Temperature; and

6. Reactor Building High Pressure.

In addition, Function 6, Reactor Building High Pressure, is required to be OPERABLE in MODE 3, whenever any CRD trip breaker is closed and the CRD system is capable of rod withdrawal. In this MODE, this Function serves purely as a back-up to other required Functions.

In MODE 1 and in MODE 2, when not in shutdown bypass operation, the following trips are required to be OPERABLE. These Functions operate to ensure that any withdrawn CONTROL RODS can be automatically inserted to make or maintain the reactor subcritical. These functions are all bypassed when the channel is placed in a shutdown bypass condition. Therefore, they are not required to be OPERABLE during shutdown bypass operation.

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE.

Two Functions are required to be OPERABLE only during portions of MODE 1. These are the Main Turbine Trip (Oil Pressure) and the Loss of Main Feedwater Pumps (Control Oil Pressure) trip. These Functions are required to be OPERABLE at ≥ 45% RTP and ≥ 10% RTP, respectively. Analyses presented in BAW-1893 (Ref. 6) have shown that for operation below these power levels, these trips are not necessary to minimize challenges to the ERV as required by NUREG-0737 (Ref. 4).

Because the safety function of the RPS is to trip the CONTROL RODS, the RPS is not required to be OPERABLE in MODE 3, 4, or 5, if either the reactor trip breakers are open, or the CRD System is incapable of rod withdrawal. Similarly, the RPS is not required to be OPERABLE in MODE 6 because the CONTROL RODS are normally decoupled from the CRDs.

However, during shutdown bypass operation, in MODE 2, 3, 4, or 5, the Shutdown Bypass RCS High Pressure and Nuclear Overpower - Low setpoint trips are required to be OPERABLE if the CRD trip breakers are closed and the CRD System is capable of rod withdrawal. Under these conditions, the Shutdown Bypass RCS High Pressure and Nuclear Overpower - Low setpoint trips sufficiently reduce the potential for conditions that could challenge SLs.

ACTIONS

Conditions A, B, and C are applicable to all RPS protection Functions. If a channel's trip setpoint is found nonconservative with respect to the required Allowable Value in Table 3.3.1-1, or the transmitter, instrument loop, signal processing electronics or bistable is found inoperable, the channel must be declared inoperable and all applicable Conditions entered immediately.

<u>A.1 and A.2</u>

If one or more Functions in one protection channel become inoperable, the affected protection channel must be placed in bypass or trip, or the bypass of the remaining channels prevented. If the channel is bypassed, all RPS Functions are placed in a two-out-of-three logic configuration and the bypass of any other channel is prevented. In this configuration, the RPS can still perform its safety function in the presence of a random failure of any single channel. Alternatively, the inoperable channel can be placed in trip. Tripping the affected protection channel places all RPS Functions in a one-out-of-three configuration.

Another option is to maintain the channel, which contains one or more inoperable Functions, in an untripped and unbypassed state. In this case, bypass of the remaining three channels must be prevented. This is accomplished by tagging them, under administrative controls, to prevent their being bypassed. This option assumes that the inoperability of the Function(s) does not require the channel containing the inoperable Function(s) to remain in a tripped condition, and that the channel contains other Functions which remain OPERABLE.

By maintaining the channel in an untripped and unbypassed state, the inoperable Function (s) are in a two-out-of-three logic configuration. This configuration is equivalent to bypassing the channel. However, by maintaining the channel in an untripped and unbypassed condition, the OPERABLE Functions within that channel remain in service in a normal two-out-of-four logic configuration.

Operation in these configurations may continue indefinitely because the RPS is capable of performing its trip Function in the presence of any single random failure. The 1 hour Completion Time is sufficient to perform Required Action A.1 or Required Action A.2.

B.1, B.2.1, and B.2.2

For Required Action B.1 and Required Action B.2, if one or more Functions in two protection channels become inoperable, one of two inoperable protection channels must be placed in trip. The second inoperable channel may be bypassed or may be maintained in an untripped and unbypassed condition. If the channel is not bypassed, bypass of the remaining channels must be prevented. This is accomplished by tagging them, under administrative controls, to prevent their being bypassed. This option assumes that the inoperability of the Function(s) in the second channel does not require that channel to remain in a tripped condition, and that the channel contains one or more Function(s) which remains OPERABLE. These Required Actions place all RPS Functions in either a one-out-of-two or one-out-of-three logic configuration. In either of these configurations, the RPS can still perform its safety functions in the presence of a random failure of any single channel. The 1 hour Completion Time is sufficient time to perform Required Action B.1, Required Action B.2.1, and Required Action B.2.2.

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.1-1. The applicable Condition referenced in the table is Function dependent. If the Required Action and associated Completion Time of Condition A or B are not met or if more than two channels are inoperable, Condition C is entered to provide for transfer to the appropriate subsequent Condition.

D.1 and D.2

If Required Action C.1 and Table 3.3.1-1 direct entry into Condition D, the unit must be brought to a MODE in which the specified RPS trip Functions are not required to be OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and to open all CRD trip breakers without challenging unit systems.

E.1

If Required Action C.1 and Table 3.3.1-1 direct entry into Condition E, the unit must be brought to a MODE in which the specified RPS trip Functions are not required to be OPERABLE. To achieve this status, all CRD trip breakers must be opened. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open CRD trip breakers without challenging unit systems.

F.1

If Required Action C.1 and Table 3.3.1-1 direct entry into Condition F, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced to < 45% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach 45% RTP from full power conditions in an orderly manner without challenging unit systems.

G.1

If Required Action C.1 and Table 3.3.1-1 direct entry into Condition G, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced to < 10% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach 10% RTP from full power conditions in an orderly manner without challenging unit systems.

SURVEILLANCE REQUIREMENTS

The SRs for each RPS Function are identified by the SRs column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION testing.

The SRs are modified by a Note which directs the reader to Table 3.3.1-1 to determine the correct SRs to perform for each RPS Function.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours provides reasonable assurance of prompt identification of a gross failure of instrumentation. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to the same parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of factors including channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are, where practical, verified to be reading at the bottom of the range and not failed downscale.

The agreement criteria includes an expectation of one decade of overlap when transitioning between neutron flux instrumentation. For example, during a power increase near the top of the scale for the intermediate range monitors, a power range monitor reading is expected with at least one decade overlap. Without such an overlap, the power range monitors are considered inoperable unless it is clear that an intermediate range monitor inoperability is responsible for the lack of the expected overlap.

The Frequency is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal but more frequent checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

For Functions that trip on a combination of several measurements, such as the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE Function, the CHANNEL CHECK must be performed on each input.

SR 3.3.1.2

This SR is the performance of a heat balance calibration for the power range channels every 96 hours and once within 24 hours after a THERMAL POWER change of ≥ 10% RTP in one direction, when reactor power is ≥ 20% RTP. The heat balance calibration consists of a comparison of the results of the calorimetric

with the power range channel output. The outputs of the power range channels are calibrated to the calorimetric. Note 1 to the SR states if the absolute difference between the calorimetric and the Nuclear Instrumentation System (NIS) channel is > 2% RTP, the NIS channel is not declared inoperable but must be adjusted. If the NIS channel cannot be properly adjusted, the channel is declared inoperable. Note 2 clarifies that this Surveillance is required only if reactor power is ≥ 20% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 20% RTP.

Two calorimetric calculations are routinely performed. One relies upon primary system parameters and the other relies upon secondary system parameters. The primary calorimetric is generally less accurate than the secondary calorimetric at higher power levels and more accurate at lower power levels. For comparison to the nuclear instrumentation, between 0 and 15% power, only the primary calorimetric (heat balance) is considered. From 15 to 100% power the calorimetric is weighted linearly with only the secondary heat balance being considered at 100% power.

The power range channel's output shall be adjusted consistent with the calorimetric results if the absolute difference between the calorimetric and the power range channel's output is > 2% RTP. The value of 2% is adequate because this value is assumed in the safety analyses of SAR, Chapter 14 (Ref. 2). These checks and, if necessary, the adjustment of the power range channels ensure that channel accuracy is maintained within the analyzed error margins. The 96 hour Frequency is adequate, based on unit operating experience, which demonstrates the change in the difference between the power range indication and the calorimetric results rarely exceeds 2% in any 96 hour period. Furthermore, the control room operators monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

A comparison of power range nuclear instrumentation channels against incore detectors shall be performed at a 31 day Frequency when reactor power is ≥ 20% RTP. The SR is modified by two Notes. Note 2 clarifies that 24 hours is allowed for performing the first Surveillance after reaching 20% RTP. Note 1 states if the absolute difference between the power range and incore AXIAL POWER IMBALANCE measurements is ≥ 2% RTP, the power range channel is not inoperable, but an adjustment of the measured imbalance to agree with the incore measurements is necessary. If the power range channel cannot be properly recalibrated, the channel is declared inoperable. The calculation of the Allowable Value envelope assumes a difference in out of core to incore AXIAL POWER IMBALANCE measurements of 2.5%. Additional inaccuracies beyond those that are measured are also included in the setpoint envelope calculation. The 31 day Frequency is adequate, considering that long term drift of the excore linear amplifiers is small and burnup of the detectors is slow. Also, the excore readings are a strong function of the power produced in the peripheral fuel bundles, and do not represent an integrated reading across the core. The slow changes in neutron flux during the fuel cycle can also be detected at this interval.

SR 3.3.1.4

A CHANNEL FUNCTIONAL TEST is performed to ensure that the entire channel will perform the intended function. Setpoints must be found within the Allowable Values specified in Table 3.3.1-1. Any setpoint adjustment shall be consistent with the assumptions of the current setpoint analysis.

The Frequency of 31 days is based on operating experience, which has demonstrated through high reliability of the instrumentation, that failure of more than one channel, of a given Function, in any 31 day interval is rare.

Testing in accordance with this SR is normally performed on a rotational basis, with one channel being tested each week. Testing one channel each week reduces the probability of an undetected failure existing within the system and minimizes the likelihood of the same systematic test errors being introduced into each redundant channel. The automatic bypass removal feature is verified for the turbine oil pressure trip and the main feedwater pump oil pressure trip functions during the CHANNEL FUNCTIONAL TEST.

SR 3.3.1.5

A Note to the Surveillance indicates that neutron detectors are excluded from CHANNEL CALIBRATION. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that instrument errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATION must be performed consistent with the assumptions of the setpoint analysis. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature (RTD) sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

The Frequency is justified by the assumption of at least an 18 month calibration interval in the determination of the allowable magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. SAR, Chapter 7.
2. SAR, Chapter 14 and Chapter 3A.
3. Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001.
4. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.
5. 10 CFR 50.36.
6. BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," October 1985.

B 3.3 INSTRUMENTATION

B 3.3.2 Reactor Protection System (RPS) Manual Reactor Trip

BASES

BACKGROUND

The RPS Manual Reactor Trip provides the operator with the capability to trip the reactor from the control room in the absence of, or coincident with, any other trip condition. Manual trip is provided by a trip push button on the main control board. This push button operates four electrically independent switches. This trip is independent of the automatic trip system. As shown in Figure 7.1, SAR, Chapter 7 (Ref. 1), control power for the control rod drive (CRD) breakers and electronic trip assembly (ETA) relays comes from the reactor trip modules (RTMs). The manual trip switches are located between the RTM output and the breaker undervoltage coils, breaker undervoltage relays, and ETA relays. The switches also initiate actuation of the breaker shunt trip mechanisms. These are separate switches which are actuated through a mechanical linkage from a single push button. Opening of the switches opens the circuits to the breakers, tripping them.

APPLICABLE SAFETY ANALYSES

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time. The Manual Reactor Trip Function is required as a backup to the automatic trip functions.

Operating experience has shown the Manual Reactor Trip Function to be significant to public health and safety, and therefore satisfy Criterion 4 of 10 CFR 50.36 (Ref. 2).

LCO

The LCO on the RPS Manual Reactor Trip requires that the trip shall be OPERABLE whenever the reactor is critical or any time any CRD breaker is closed and rods are capable of being withdrawn, including shutdown bypass. This enables the operator to terminate any reactivity excursion that in the operator's judgment requires protective action, even if no automatic trip condition exists.

The Manual Reactor Trip Function is composed of four electrically independent trip switches sharing a common mechanical push button.

APPLICABILITY

The Manual Reactor Trip Function is required to be OPERABLE in MODES 1 and 2. It is also required to be OPERABLE in MODES 3, 4, and 5 if any CRD trip breaker is in the closed position and if the CRD System is capable of rod withdrawal. The primary safety function of the RPS is to trip the CONTROL RODS; therefore, the Manual Reactor Trip Function is not needed in MODE 3, 4, or 5 if the reactor trip breakers are open or if the CRD System is incapable of rod withdrawal. Similarly, the RPS Manual Reactor Trip is not needed in MODE 6 because the CONTROL RODS are normally decoupled from the CRDs.

ACTIONS

A.1

Condition A applies when the Manual Reactor Trip Function is found inoperable. One hour is allowed to restore the Function to OPERABLE status. The automatic functions and various alternative manual trip methods, such as removing power to the RTMs, are still available. The 1 hour Completion Time is sufficient time to correct minor problems.

B.1 and B.2

If the Required Action and associated Completion Time are not met in MODE 1, 2, or 3, the unit must be placed in a MODE in which manual trip is not required. Required Action B.1 and Required Action B.2 place the unit in at least MODE 3 with all CRD trip breakers open within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

C.1

If the Required Action and associated Completion Time are not met in MODE 4 or 5, the unit must be placed in a MODE in which manual trip is not required. To achieve this status, all CRD trip breakers must be opened. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the Manual Reactor Trip Function. This test verifies the OPERABILITY of the Manual Reactor Trip by actuation of the CRD trip breakers. The Frequency shall be once prior to each reactor startup if not performed within the preceding 7 days to ensure the OPERABILITY of the Manual Reactor Trip Function prior to achieving criticality. The Frequency was developed in consideration that this Surveillance is only performed during a unit outage.

REFERENCES

1. SAR, Chapter 7.
2. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.3 Reactor Protection System (RPS) - Reactor Trip Module (RTM)

BASES

BACKGROUND

The RPS consists of four independent protection channels, each containing an RTM. Figure 7.1, SAR, Chapter 7 (Ref. 1), shows a typical RPS protection channel and the relationship of the RTM to the RPS instrumentation, manual trip, and control rod drive (CRD) trip devices. The RTM receives bistable trip signals from the functions in its own channel and channel trip signals from the other three RPS - RTMs. The RTM provides these signals to its own two-out-of-four trip logic and transmits its own channel trip signal to the two-out-of-four logic of the RTMs in the other three RPS channels. Whenever any two RPS channels transmit channel trip signals, the RTM logic in each channel actuates to remove 120 VAC power from its associated CRD trip device.

The RPS trip scheme consists of series contacts that are operated by bistables. During normal unit operations, all contacts are closed and the RTM channel trip relay remains energized. However, if any trip parameter exceeds its setpoint, its associated contact opens, which de-energizes the channel trip relay.

When an RTM channel trip relay de-energizes, several things occur:

a. Each of the four (4) output logic relays "informs" its associated RPS channel that a reactor trip signal has occurred in the tripped RPS channel;

b. The contacts in the trip device circuitry, powered by the tripped channel, open, but the trip device remains energized through the closed contacts from the other RTMs. (This condition exists in each RPS - RTM. Each RPS - RTM controls power to a trip device.); and

c. The contact in parallel with the channel reset switch opens and the trip is sealed in. To re-energize the channel trip relay, the channel reset switch must be depressed after the trip condition has cleared.

When the second RPS channel senses a reactor trip condition, the output logic relays for the second channel de-energize and open contacts that supply power to the trip devices. With contacts opened by two separate RPS channels, power to the trip devices is interrupted and the CONTROL RODS fall into the core.

A minimum of two out of four RTMs must sense a trip condition to cause a reactor trip. Also, two channel trips caused by different trip functions can result in a reactor trip.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on a reactor trip for protection of reactor core integrity, reactor coolant pressure boundary integrity, and reactor building OPERABILITY. A reactor trip must occur when needed to prevent accident conditions from exceeding those calculated in the accident analyses. More detailed descriptions of the applicable accident analyses are found in the bases for each of the RPS trip Functions in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation."

In MODES 1 and 2, the RTMs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2). In MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal, the RTMs satisfy Criterion 4 of 10 CFR 50.36.

LCO

The RTM LCO requires all four RTMs to be OPERABLE. Failure of any RTM renders a portion of the RPS inoperable.

To be considered OPERABLE, an RTM must be able to receive and interpret trip signals from its own and other OPERABLE RPS channels and to open its associated trip device.

The requirement for four channels to be OPERABLE ensures that a minimum of two RPS channels will remain OPERABLE if a single failure has occurred in one channel and if a second channel has been bypassed. This two-out-of-four trip logic also ensures that a single RPS channel failure will not cause an unwanted reactor trip. Violation of this LCO could result in a trip signal not causing a reactor trip when needed.

APPLICABILITY

The RTMs are required to be OPERABLE in MODES 1 and 2. They are also required to be OPERABLE in MODES 3, 4, and 5 if any CRD trip breakers are in the closed position and the CRD System is capable of rod withdrawal. The RTMs are designed to ensure a reactor trip would occur, if needed. This need may exist in any of these MODES; therefore, the RTMs must be OPERABLE.

ACTIONS

A.1.1, A.1.2, and A.2

When an RTM is inoperable, the associated CRD trip breaker must then be placed in a condition that is equivalent to a tripped condition for the RTM. Required Action A.1.1 or Required Action A.1.2 requires this either by opening (tripping) the CRD trip breaker or by removing power to the CRD trip device. Tripping one RTM

or removing power opens one set of CRD trip devices. Power to hold up CONTROL RODS is still provided via the parallel CRD trip device(s). Therefore, a reactor trip will not occur until a second protection channel trips.

To ensure the trip signal is registered in the other channels, Required Action A.2 requires that the inoperable RTM be removed from the cabinet. This action causes the electrical interlocks to indicate a tripped channel in the remaining three RTMs. Operation in this condition is allowed indefinitely because the actions put the RPS into a one-out-of-three configuration. The 1 hour Completion Time is sufficient time to perform the Required Actions.

B.1, B.2.1, and B.2.2

Condition B applies if two or more RTMs are inoperable in MODE 1, 2, or 3, or if the Required Actions and associated Completion Time of Condition A are not met in MODE 1, 2, or 3. In this case, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 with all CRD trip breakers open or with power to all CRD trip breakers removed within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

C.1 and C.2

Condition C applies if two or more RTMs are inoperable in MODE 4 or 5, or if the Required Actions and associated Completion Times are not met in MODE 4 or 5. In this case, the unit must be placed in a MODE in which the LCO does not apply. This is done by opening all CRD trip breakers or removing power from all CRD trip breakers. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove all power to the CRD System without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

The SRs include performance of a CHANNEL FUNCTIONAL TEST every 92 days. This test shall verify the OPERABILITY of the RTM and its ability to receive and properly respond to channel trip and reactor trip signals.

The Frequency of 92 days is based on operating experience, which has demonstrated through high reliability of the instrumentation, that failure of more than one RTM in any 92 day interval is rare (Ref. 3).

Testing in accordance with this SR is normally performed on a rotational basis, with one RTM being tested each 23 days. Testing one RTM each 23 days reduces the probability of an undetected failure existing within the system and minimizes the likelihood of the same systematic test errors being introduced into each redundant RTM.

REFERENCES

1. SAR, Chapter 7.

2. 10 CFR 50.36.

3. BAW-10167A, "Justification for Increasing the Reactor Trip System On-Line Test Intervals," Supplement 3, "Justification for Increasing the Trip Device Test Interval," February 1998.

B 3.3 INSTRUMENTATION

B 3.3.4 Control Rod Drive (CRD) Trip Devices

BASES

BACKGROUND

The Reactor Protection System (RPS) contains multiple CRD trip devices: two AC trip breakers, two DC trip breaker pairs, and ten electronic trip assembly (ETA) relays. The system has two separate paths (or channels), with each path having one AC breaker either in series with a pair of DC breakers or functionally in series with five ETA relays in parallel. Each path provides independent power to the CRDs. Either path can provide sufficient power to operate the entire CRD System.

Figure 7-10 SAR, Chapter 7 (Ref. 1), illustrates the configuration of CRD trip devices. To trip the reactor, power to the CRDs must be removed. Loss of power causes the CRD's mechanisms to release the CONTROL RODS, which then fall by gravity into the core.

Power to CRDs is supplied from two separate unit sources through the AC trip circuit breakers. These breakers are designated A and B, and their undervoltage and shunt trip coils are controlled by RPS channels A and B, respectively. From the circuit breakers, the CRD power travels through voltage regulators and stepdown transformers. These devices in turn supply redundant buses that feed the DC holding power supplies and the regulating rod, APSR and auxiliary power supplies.

The DC holding power supplies rectify the AC input and supply power to hold the safety rods in their fully withdrawn position. One of the redundant power sources supplies phase A; the other, phase CC. Either phase being energized is sufficient to hold the rod. Two breakers are located on the output of each power supply. Each breaker controls half of the power to two of the four safety rod groups. The undervoltage and shunt trip coils on the two circuit breakers on the output of one of the power supplies is controlled by RPS channel C. The other two breakers are controlled by RPS channel D.

In addition to the DC holding power supplies, the redundant buses also supply power to the regulating rod, APSR and auxiliary power supplies. These power supplies contain silicon controlled rectifiers (SCRs), which are gated on and off to provide power to, and remove power from, the phases of the CRD mechanisms. The gating control signal for these SCRs is supplied through the closed contacts of the ETA relays. These contacts are referred to as E and F contactors, and are controlled by the C and D RPS channels, respectively.

The AC breaker and DC breakers, or gated SCRs, are in series in one of the power supplies; whereas, the redundant AC breaker and DC breakers or gated SCRs are in series in the other power supply to the CONTROL RODS. The logic required to cause a reactor trip is the opening of a circuit breaker or ETA relay in each of the

redundant power supplies. (The pair of DC circuit breakers on the output of the power supply are treated as one breaker.) This is known as a one-out-of-two taken twice logic. The following examples illustrate the operation of the reactor trip circuit breakers.

a. If the A AC circuit breaker opens:

   1. the input power to associated DC power supply is lost, and

   2. the SCR supply from the associated power source is lost.

b. If the D DC circuit breaker(s) and F contactors open:

   1. the output of the redundant DC power supply is lost and the safety rods de-energize, and

   2. when the F contactor opens, SCR gating power is lost and the regulating rods will be de-energized.

c. The combination of (a) and (b) causes a reactor trip.

Any other combination of at least one circuit breaker opening in each power supply will cause a reactor trip.

In summary, two tripped RPS channels will cause a reactor trip. For example, a reactor trip occurs if RPS channel B senses a low Reactor Coolant System (RCS) pressure condition and if RPS channel C senses a variable low RCS pressure condition. When the channel B bistable relay de-energizes, the channel trip relay de-energizes and opens its associated contacts. The same thing occurs in channel C, except the variable low pressure bistable relay de-energizes the channel C trip relay. When the output logic relays in channels B and C de-energize, the B and C contacts in the trip logic of each channel's reactor trip module (RTM) open causing an undervoltage to each trip breaker. All trip breakers and the ETA relay contactors open, and power is removed from all CRD mechanisms. All rods fall into the core, resulting in a reactor trip.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on a reactor trip for protection of reactor core integrity, reactor coolant pressure boundary integrity, and reactor building OPERABILITY. A reactor trip must occur when needed to prevent accident consequences from exceeding those calculated in the accident analyses. The control rod insertion limits ensure that adequate rod worth is available upon reactor trip to shut down the reactor to the required SDM. Further, OPERABILITY of the CRD trip devices ensures that all CONTROL RODS will trip when required. More detailed descriptions of the applicable accident analyses are found in the Bases for each of the individual RPS trip Functions in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation."

In MODES 1 and 2, the CRD trip devices satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2). In MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal, the CRD trip devices satisfy Criterion 4 of 10 CFR 50.36.

LCO

The LCO requires all of the specified CRD trip devices to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when any CRD trip breaker is in the closed position and the CRD System is capable of rod withdrawal. Failure of any required CRD trip device renders a portion of the RPS inoperable. Without reliable CRD reactor trip circuit breakers and associated support circuitry, a reactor trip may not occur when initiated either automatically or manually.

All required CRD trip devices shall be OPERABLE to ensure that the reactor remains capable of being tripped any time it is critical. OPERABILITY is defined as the CRD trip device being able to receive a reactor trip signal and to respond to this trip signal by interrupting power to the CRDs. Both of a CRD trip breaker's diverse trip devices and the breaker itself must be functioning properly for the breaker to be OPERABLE.

Both ETA relays associated with each of the three regulating rod groups and the two ETA relays associated with the auxiliary power supply must be OPERABLE to satisfy the LCO. The ETA relays associated with the APSR power supply are not required to be OPERABLE because the APSRs are not designed to fall into the core upon initiation of a reactor trip.

Requiring all breakers and ETA relays to be OPERABLE ensures that at least one device in each of the two power paths to the CRDs will remain OPERABLE even with a single failure.

APPLICABILITY

The CRD trip devices are required to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when any CRD trip breaker is in the closed position and the CRD System is capable of rod withdrawal.

The CRD trip devices are designed to ensure that a reactor trip would occur if needed. Since a trip may be required in all of these MODES, the CRD trip devices must be OPERABLE.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each CRD trip device.

A.1 and A.2

Condition A represents reduced redundancy in the CRD trip Function. Condition A applies when:

- One diverse trip Function (undervoltage or shunt trip device) is inoperable in one or more CRD trip breaker(s) or breaker pair; or

- One diverse trip Function is inoperable in both DC trip breakers associated with one protection channel. In this case, the inoperable trip Function does not need to be the same for both breakers.

If one of the diverse trip Functions on a CRD trip breaker or breaker pair becomes inoperable, actions must be taken to preclude the inoperable CRD trip device from preventing a reactor trip when needed. This is done by manually opening the inoperable CRD trip breaker or by removing power from the inoperable CRD trip breaker. Either of these actions places the affected CRDs in a one-out-of-two trip configuration, which precludes a single failure, which in turn could prevent tripping of the reactor. The 48 hour Completion Time has been shown to be acceptable through operating experience.

B.1 and B.2

Condition B represents a loss of redundancy for the CRD trip Function. Condition B applies when both diverse trip Functions are inoperable in one or more trip breaker(s) or breaker pairs.

Required Action B.1 and Required Action B.2 are the same as Required Action A.1 and Required Action A.2, but the Completion Time is shortened. The 1 hour Completion Time allowed to open or remove power from the CRD trip breaker allows the operator to take all the appropriate actions for the inoperable breaker and still ensures that the risk involved is acceptable.

C.1, C.2, C.3, and C.4

Condition C represents a loss of redundancy for the CRD trip Function. Condition C applies when one or more ETA relays are inoperable. The preferred action is to restore the ETA relay to OPERABLE status. If this cannot be done, the operator can perform one of four actions to eliminate reliance on the failed ETA relay. The first option is to switch the affected CONTROL ROD group to an alternate power supply which has two OPERABLE or one OPERABLE and one open ETA relay. This removes the failed ETA relay from the trip sequence, and the unit can operate indefinitely. The second option is to transfer the affected CONTROL ROD group to a DC holding power supply. This option is only available if the affected group is a safety rod group and the affected power supply is the auxiliary power supply. The third option is to open the inoperable ETA contacts. This option results in the safety function being performed. The fourth option is to open the corresponding AC CRD

trip breaker. This also results in the safety function being performed, thereby eliminating the failed ETA relay from the trip sequence.

The 1 hour Completion Time is sufficient to perform the Required Action.

D.1, D.2.1, and D.2.2

If the Required Actions and associated Completion Times of Condition A, B, or C are not met in MODE 1, 2, or 3, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3, with all CRD trip breakers open or with power to all CRD trip breakers removed within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

E.1 and E.2

If the Required Actions and associated Completion Times of Condition A, B, or C are not met in MODE 4 or 5, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, all CRD trip breakers must be opened or power to all CRD trip breakers removed within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove power from all CRD trip breakers without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

SR 3.3.4.1 is to perform a CHANNEL FUNCTIONAL TEST every 92 days. This test verifies the OPERABILITY of the trip devices by actuation of the end devices. Also, this test independently verifies the undervoltage and shunt trip mechanisms of the trip breakers. The Frequency of 92 days is based on operating experience, which has demonstrated that failure of more than one channel of a given function in any 92 day interval is a rare event (Ref. 3).

Testing in accordance with this SR is normally performed on a rotational basis with one channel being tested each 23 days. Testing one channel each 23 days reduces the probability of an undetected failure existing within the system and minimizes the likelihood of the same systematic test errors being introduced into each redundant trip device.

REFERENCES

1. SAR, Chapter 7.

2. 10 CFR 50.36.

3. BAW-10167A, "Justification for Increasing the Reactor Trip System On-Line Test Intervals," Supplement 3, "Justification for Increasing the Trip Device Test Interval," February 1998.

B 3.3 INSTRUMENTATION

B 3.3.9 Source Range Neutron Flux

BASES

BACKGROUND

The source range neutron flux channels provide the operator with an indication of the approach to criticality. These channels also provide the operator with a flux indication that reveals changes in reactivity.

The source range instrumentation has two redundant count rate channels originating in two high sensitivity fission chambers. Two source range detectors are externally located on opposite sides of the core. These channels are used over a counting range of 0.1 cps to 1E5 cps and are displayed on the operator's control console in terms of log count rate. The channels also measure the rate of change of the neutron flux level, which is displayed for the operator in terms of startup rate from -1 decades to +7 decades per minute. An interlock provides a control rod withdraw "inhibit" on a high startup rate of +2 decades per minute in either channel. This interlock is bypassed when the intermediate range neutron flux channels reach 1E-9 amps or power range neutron flux channels reach 10% RTP.

APPLICABLE SAFETY ANALYSES

The source range neutron flux channels are necessary to monitor core reactivity changes. They are the primary means for detecting reactivity changes and triggering operator actions to respond to reactivity transients initiated from conditions in which the Reactor Protection System (RPS) is not required to be OPERABLE. They also trigger operator actions to anticipate RPS actuation in the event of reactivity transients starting from shutdown or low power conditions.

The source range neutron flux channels satisfy Criterion 4 of 10 CFR 50.36 (Ref. 1).

LCO

One source range neutron flux channel shall be OPERABLE to provide the operator with source range neutron instrumentation. The source range instrumentation provides the primary power indication at ≤ 1E-10 amp on intermediate range instrumentation and must remain OPERABLE for the operator to continue increasing power.

APPLICABILITY

One source range neutron flux channel shall be OPERABLE in MODE 2 to provide indication during an approach to criticality. Neutron flux level is sufficient for monitoring on the intermediate range and on the power range instrumentation prior to entering MODE 1; therefore, source range instrumentation is not required in MODE 1.

In MODES 3, 4, and 5, source range neutron flux instrumentation shall be OPERABLE to provide the operator with a means of monitoring neutron flux and to provide an early indication of reactivity changes.

The requirements for source range neutron flux instrumentation during MODE 6 refueling operations are addressed in LCO 3.9.2, "Nuclear Instrumentation."

ACTIONS

A.1, A.2, A.3, and A.4

With the required source range neutron flux channel inoperable with ≤ 1E-10 amp on the intermediate range neutron flux instrumentation, the operators must take actions to limit the possibilities for adding positive reactivity. This is done by immediately suspending positive reactivity additions, initiating action to insert all CONTROL RODS, and opening the control rod drive trip breakers within 1 hour. RCS temperature changes are permitted, however, provided the effects of such temperature changes are accounted for in the SDM calculations. Periodic SDM verification is then required to provide a means for detecting the slow reactivity changes that could be caused by mechanisms other than CONTROL ROD withdrawal or operations involving positive reactivity changes. Since the source range instrumentation provides the only reliable direct indication of power in these MODES, the operators must continue to verify the SDM every 12 hours until at least one channel of the source range instrumentation is returned to OPERABLE status. Required Action A.1, Required Action A.2, and Required Action A.3 preclude rapid positive reactivity additions. The 1 hour Completion Time for Required Action A.3 and Required Action A.4 provides sufficient time for operators to accomplish the actions. The 12 hour Frequency for performing the SDM verification provides reasonable assurance that the reactivity changes possible with CONTROL RODS inserted are detected before SDM limits are challenged.

If no indication of intermediate range flux is available, these Required Actions are also appropriate.

B.1

With > 1E-10 amp in MODE 2, 3, 4, or 5 on the intermediate range neutron flux instrumentation, continued operation is allowed with the required source range

neutron flux channel inoperable. The ability to continue operation is justified because the instrumentation does not provide a safety function during high power operation. However, actions are initiated within 1 hour to restore the required channel to OPERABLE status for future availability. The Completion Time of 1 hour is sufficient to initiate the action. The action must continue until the channel is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

Performance of the CHANNEL CHECK once every 12 hours provides reasonable assurance of prompt identification of a gross failure of instrumentation. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to the same parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of factors including channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction.

The agreement criteria includes an expectation of one decade of overlap when transitioning between neutron flux instrumentation. For example, during a power reduction near the bottom of the scale for the intermediate range monitors, a source range monitor reading is expected with at least one decade overlap. Without such an overlap, the source range monitors are considered inoperable unless it is clear that an intermediate range monitor inoperability is responsible for the lack of the expected overlap.

The Frequency is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channel. When operating with only one channel OPERABLE, CHANNEL CHECK is still required. However, in this condition, a redundant source range may not be available for comparison. CHANNEL CHECK may still be

performed via comparison with intermediate range detectors, if available, and verification that the OPERABLE source range channel is energized and indicating a value consistent with current unit status.

SR 3.3.9.2

For a source range neutron flux channel, CHANNEL CALIBRATION is a complete check and readjustment of the channel from the preamplifier input to the indicator. This test verifies the channel responds to measured parameters within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel at a setpoint which accounts for instrument drift to ensure that the instrument channel remains operational between successive tests.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult, and there is no adjustment that can be made to the detectors. Furthermore, adjustment of the detectors is unnecessary because they are passive devices with minimal drift. Finally, the detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output.

The Frequency of 18 months is based on demonstrated instrument CHANNEL CALIBRATION reliability over an 18 month interval, such that the instrument is not adversely affected by drift.

REFERENCES

1. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.10 Intermediate Range Neutron Flux

BASES

BACKGROUND

The intermediate range neutron flux channels provide the operator with an indication of reactor power at higher power levels than the source range instrumentation and lower power levels than the power range instrumentation.

The intermediate range instrumentation has two channels originating in two gamma compensated ion chambers. Each channel provides eight decades of flux level information in terms of the log of ion chamber current from 1E-11 amp to 1E-3 amp. The channels also measure the rate of change of the neutron flux level, which is displayed for the operator in terms of startup rate from -0.5 decades to +5 decades per minute. A high startup rate of +3 decades per minute in either channel will initiate a control rod withdrawal inhibit while below 10% RTP.

The intermediate range compensated ion chambers are of the electrically adjustable gamma compensating type. Each detector has a separate adjustable high voltage power supply and an adjustable compensating voltage supply.

APPLICABLE SAFETY ANALYSES

Intermediate range neutron flux channels are necessary to monitor core reactivity changes and provide the primary indication to trigger operator actions to anticipate Reactor Protection System actuation in the event of reactivity transients starting from low power conditions.

The intermediate range neutron flux channels satisfy Criterion 4 of 10 CFR 50.36 (Ref. 1).

LCO

One intermediate range neutron flux instrumentation channel shall be OPERABLE to provide the operator with neutron flux indication. This enables operators to control the increase in power and to detect neutron flux transients. This indication is used until the power range instrumentation is on scale. Violation of this requirement could prevent the operator from detecting and controlling neutron flux transients that could result in reactor trip during power escalation.

APPLICABILITY

The required intermediate range neutron flux channel shall be OPERABLE in MODE 2 and in MODES 3, 4, and 5 with any control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

The intermediate range instrumentation is designed to detect power changes when the power range and source range instrumentation cannot provide reliable indications, e.g., during initial criticality and power escalation. Since those conditions can exist in, or propagate from, all of these MODES, the intermediate range instrumentation must be OPERABLE.

ACTIONS

A.1 and A.2

With the required intermediate range neutron flux channel inoperable when THERMAL POWER is ≤ 5% RTP, the operators must place the reactor in the next lowest condition for which the intermediate range instrumentation is not required. This involves providing power level indication on the source range instrumentation by immediately suspending operations involving positive reactivity changes and, within 1 hour, placing the reactor in the tripped condition with the CRD trip breakers open. RCS temperature changes are permitted provided the effects of such changes are accounted for in the SDM caculations. The Completion Times are based on unit operating experience and allow the operators sufficient time to manually insert the CONTROL RODS prior to opening the CRD breakers.

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

Performance of the CHANNEL CHECK once every 12 hours provides reasonable assurance of prompt identification that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to the same parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of factors including channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that the signal processing equipment has drifted

outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. Off scale low current loop channels are verified, where practical to be reading at the bottom of the range and not failed low.

The agreement criteria includes an expectation of one decade of overlap when transitioning between neutron flux instrumentation. For example, during a power increase near the top of the scale for the source range monitors, an intermediate range monitor reading is expected with at least one decade overlap. Without such an overlap, the intermediate range monitors are considered inoperable unless it is clear that a source range monitor inoperability is responsible for the lack of the expected overlap. Further, during a power reduction near the bottom of the scale for the power range monitors, an intermediate range monitor reading is expected with at least one decade overlap. Without such an overlap, the intermediate range monitors are considered inoperable unless it is clear that a power range monitor inoperability is responsible for the lack of the expected overlap.

The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channel.

When operating with only one channel OPERABLE, CHANNEL CHECK is still required. However, in this condition, a redundant intermediate range is not available for comparison. CHANNEL CHECK may still be performed via comparison with power or source range detectors, if available, and verification that the OPERABLE intermediate range channel is energized and indicates a value consistent with current unit status.

<u>SR 3.3.10.2</u>

A CHANNEL FUNCTIONAL TEST, of the required intermediate range instrument channel, verifies proper operation of the channel each 31 days. Monthly testing provides reasonable assurance that the instrument channel will function, if required, to provide indication during MODE 2 and during unanticipated reactivity excursions from MODES 3, 4, or 5.

<u>SR 3.3.10.3</u>

For intermediate range neutron flux channels, CHANNEL CALIBRATION is a complete check and readjustment of the channels, from the preamplifier input to the indicators. This test verifies the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel at a setpoint which accounts for instrument drift to ensure that the instrument channel remains operational between successive tests.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. In addition, the detectors are of simple construction, and any failures in the detectors will be apparent as a change in channel output. The Frequency is based on operating experience and consistency with the typical industry refueling cycle and is justified by demonstrated instrument reliability over an 18 month interval such that the instrument is not adversely affected by drift.

REFERENCES

1. 10 CFR 50.36.

CTS DISCUSSION OF CHANGES

ITS Section 3.3A: Instrumentation - RPS

Note: ITS Section 3.3A package includes the following ITS:

ITS 3.3.1	Reactor Protection System (RPS) Instrumentation
ITS 3.3.2	RPS Manual Reactor Trip
ITS 3.3.3	RPS—Reactor Trip Module (RTM)
ITS 3.3.4	Control Rod Drive (CRD) Trip Devices
ITS 3.3.9	Source Range Neutron Flux
ITS 3.3.10	Intermediate Range Neutron Flux

which address the corresponding NUREG-1430 RSTS.

ADMINISTRATIVE

A1 The designated change represents a non-technical, non-intent change to the Arkansas Nuclear One, Unit 1 Current Technical Specifications (CTS) made to make the ANO-1 Improved Technical Specifications (ITS) consistent with the B&W Standard Technical Specification (RSTS), NUREG 1430, Revision 1. This change does not alter the requirements of the CTS or RSTS. Examples of this type of change include: wording preference; convention adoption; editorial, numbering and formatting changes; and hierarchy structure.

A2 The ANO-1 CTS Bases will be administratively deleted in their entirety in favor of the NUREG-1430 RSTS Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 CTS 3.5.1.1, 3.5.1.2, and 3.5.1.3 represent information on the proper action when the number of channels is less than required by CTS Table 3.5.1-1. For example, CTS 3.5.1 does not clearly specify that the number of channels identified in Table 3.5.1-1, Column 1, are required to be OPERABLE, but CTS 3.5.1.3 provides limitations for any one or two channels inoperable. Similarly, CTS Specifications 4.1.a, 4.1.b, and 4.1.c contain information on the proper application of CTS Table 4.1-1. These Specifications and the format of the referenced Tables are replaced with the appropriate ITS requirements. The CTS markup for these Specifications and Tables does not attempt to depict all of the changes required to adopt the ITS format. Rather, the appropriate specific Discussion of Change (DOC) is indicated along with the appropriate CTS versus ITS cross reference. Therefore, this change in format is considered administrative.

A4 Surveillance frequencies in CTS Table 4.1-1 have been replaced with those from NUREG-1430. The CTS and corresponding ITS Frequencies are as follows:

CTS	ITS
S - Each shift	12 hours
W - Weekly	7 days
M - Monthly	31 days
D - Daily	24 hours
T/W - Twice per week	96 hours (See DOC L2 and M4)
Q - Quarterly	92 days
P - Prior to startup if not done previous week	Not Used
B/M - Every 2 months	Not Used
R - Once every 18 months	18 months
PC - Prior to going Critical if not done within previous 31 days	Not Used
NA - Not Applicable	Not Used
SA - SA Twice per Year	184 days

Each of these changes is consistent with the current application of the CTS frequencies at ANO-1. These changes maintain requirements consistent with both CTS and NUREG-1430. These changes are administrative in nature because they represent a change in presentation format only with no change of actual requirements.

A5 The CTS requirement to perform heat balance calibrations "daily under non-steady state operating conditions" has been retained in ITS in the Frequency of SR 3.3.1.2. This portion of the ITS Frequency is stated as "Once within 24 hours after a THERMAL POWER change of ≥ 10% RTP in one direction." This change in wording provides requirements consistent with the ANO-1 application of the CTS requirements. This change provides a change in format and a clarification of existing requirements. No new requirements are added and no existing requirements are removed by this change.

A6 The power range nuclear instruments at ANO-1 provide the nuclear power input to the Reactor Protection System (RPS) channels. This input is used in several reactor trip functions including the nuclear overpower trip function. CTS Table 3.5.1-1 and Table 4.1-1 provide specific requirements for these power range instrument channels, as well as requirements for all but one of the reactor trip functions which depend on these instruments for input. The nuclear overpower trip function is not specifically addressed in CTS Table 3.5.1-1 or Table 4.1-1. The power range nuclear instrument channels will not be specifically dealt with in ITS. Rather, ITS Specification 3.3.1 including Table 3.3.1-1 will deal individually with each RPS trip function which receives input from the power range instrument channels, including the nuclear overpower trip function. This change is administrative in nature because it represents a change, in the presentation of these requirements, with no actual change in requirements.

CTS DISCUSSION OF CHANGES

A7 The term Minimum Degree of Redundancy as presented in CTS, i.e., Table 3.5.1-1 Column 4, will not be retained in ITS. Omission of this term is not considered to result in any changes in requirements since the intent of this column is consistent with application of Table 3.5.1-1 Column 3, "Minimum Channels Operable," which is retained (although the format is changed per DOC A3). Removal of this term and its usage from the CTS does not represent any actual change in requirements, only a change in presentation.

A8 CTS Table 3.5.1-1 RPS Functional Unit 13 "Electronic (SCR) Trip Relay" indicates that there are two channels of electronic trip relays and that they must both be OPERABLE. Each of these channels consists of five electronic trip relays, four of which are required. The four required relays are associated with the three regulating rod power supplies and the auxiliary power supply which can be manually selected to power rods from any of the eight groups. The two which are not required are associated with the power supply for the AXIAL POWER SHAPING RODS (APSRs). OPERABILITY of the relays in the APSR power supply are not required because these rods are designed not to insert upon a reactor trip. The actions associated with these CTS requirements, which are found in Table 3.5.1-1 Note 23, deal specifically with one inoperable relay and with two or more inoperable relays.

NUREG-1430 treats each of these required relays individually by specifying in LCO 3.3.4 that eight "electronic trip assembly (ETA) relays" must be OPERABLE. (Note: CTS term "Electronic (SCR) Trip Relay" is considered equivalent to ITS term "electronic trip assembly (ETA) relays.") The change from specifying two channels, each of which by design contains four required relays, to specifying eight individual relays is administrative in nature. No new requirements are added by this change nor are any existing requirements removed by it. This change provides requirements consistent with NUREG-1430.

A9 NUREG-1430 3.3.4 ACTIONS NOTE has been adopted in ITS. This note allows separate Condition entry for each CRD trip device. The adoption of this Note maintains flexibility similar to that provided by the CTS. The CTS allows application of the action requirements found in Table 3.5.1-1 Notes 24 and 25 separately to each type of CRD trip breakers (AC and DC). This flexibility is retained in the ITS by application of the ACTIONS NOTE to 3.3.4 Conditions A and B. CTS Table 3.5.1-1 Note 23 provides specific action requirements for two or more inoperable Electronic Trip Relays. Application of the ACTIONS NOTE to 3.3.4 Condition C retains these requirements. No new requirements are added and no existing requirements deleted by this change.

A10 The Note modifying ITS SR 3.3.1.6 has been adopted. This Note specifically excludes neutron detectors from CHANNEL CALIBRATIONS. While the allowance provided by this Note was not specifically expressed in CTS, the application of the ANO-1 CTS definition of Instrument Channel Calibration has in practice excluded the neutron detectors. This exclusion has been made due to the passive design of the detectors, the extreme difficulty in both accessing the detectors and in generating an appropriate input signal to the detectors, and the fact that no specific adjustments can be made to the detectors. Although no specific exceptions, as allowed by this Note, exist in CTS, its adoption is administrative in nature because no actual change in requirements is made by adopting it.

A11 The CTS Table 4.1-1 Item 4 specifies the testing requirements for the Power Range Channel and CTS Table 4.1-1 Item 10 specifies the testing requirements for the Flux-Reactor Coolant Flow Comparator. These testing requirements have been retained in ITS Table 3.3.1-1 and are specifically applied to each appropriate Function in ITS Table 3.3.1-1.

As applied at ANO-1, CTS Table 4.1-1 requires a monthly CHANNEL CALIBRATION and monthly CHANNEL FUNCTIONAL TEST of both the Nuclear Overpower Function and the Nuclear Overpower RCS Flow and Measured AXIAL IMBALANCE Function. Both the CTS and ITS definitions specify that the required calibration includes the CHANNEL FUNCTIONAL TEST. Therefore, the specific requirements to perform CHANNEL FUNCTIONAL TESTS on the same Frequency as the CHANNEL CALIBRATIONS are not retained in the ITS. This change represents no actual change in requirements, only a change in presentation of requirements.

A12 CTS Table 2.3-1 indicates the Reactor Protection System Trip Setting Limit for the high reactor building pressure trip function as a maximum of 4 psig and parenthetically indicates that this is equivalent to 18.7 psia. This limit has been incorporated into the ITS as the Allowable Value for the Reactor Building High Pressure Function in Table 3.3.1-1. This value has been specified as ≤18.7 psia to be consistent with the actual design of the instrumentation used for this function. The removal of the reference to the equivalent value of 4 psig is administrative in nature and represents no change in requirements, only a change in presentation.

A13 The allowances and requirements of CTS Table 3.5.1-1 Note 24 part "b." are not specifically retained in ITS. (Note that the word "operable" in the first sentence of Note 24 part "b." was inadvertently changed from "inoperable," following the original insertion of this Note into CTS.) Based on the history of CTS, Table 3.5.1-1 Note 24 appears to have been adopted from similar requirements found in NUREG-0103 Rev. 4, Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors, Table 3.3-1 Action 7. Part "b." of CTS Table 3.5.1-1 Note 24, and NUREG-0103 Table 3.3-1 Action 7 both provide allowances to bypass a channel, to allow testing, while operating with one channel inoperable. The design of the CRD trip breakers, at ANO-1, does not contain a bypass feature. Therefore, the allowance provided by CTS Table 3.5.1-1, Note 24, part "b.," is not appropriate for the CRD trip breakers and cannot be implemented at ANO-1.

The removal of the allowances of CTS Table 3.5.1-1, Note 24, part "b." is administrative in nature because, due to the design of the equipment to which they are applicable, these allowances cannot be used. Removal of these requirements will result in no actual change in the application of CTS Table 3.5.1-1 Note 24 requirements.

A14 Not used.

A15 Not used.

A16 Not used.

A17 Three channels of power range instruments within RPS are required by CTS Table 3.5.1-1. Note 6 modifies the requirement to allow a reduction to two OPERABLE channels indefinitely, provided one of the inoperable channels is placed in the tripped condition and the other inoperable channel is placed in the bypassed condition. This requirement is further modified by Note 4. For example, if it is desired to not place one of the inoperable channels in the tripped condition, Note 4 would allow this configuration for up to four hours to allow for testing, calibration or maintenance. This 4-hour allowance is provided assuming the degree of redundancy is maintained at one. This requirement will be replaced by the Nuclear Overpower High Setpoint specified in ITS Table 3.3.1-1 and the Conditions of ITS 3.3.1. The change will retain the requirement to have a minimum of two OPERABLE channels of power range instrumentation. Operation with only two OPERABLE channels will be allowed to continue indefinitely provided the requirements of ITS 3.3.1 Condition B are met. The Required Actions of Condition B place the system in the same configuration as required by CTS Table 3.5.1-1 Note 6. By tripping one inoperable channel and bypassing the other, the system is reduced to a one out of two logic for the remaining OPERABLE channels. The exception allowed by Note 4 is retained within ITS 3.0.5 which allows returning an inoperable channel to service under certain conditions for the purposes of testing redundant channels. This change provides requirements consistent with NUREG-1430 and the current license bases and provides consistent requirements for each of the trip functions within RPS.

3.3.1-03

3.3.1-04

A18 The Frequency for performance of heat balance calibrations was changed. The CTS twice weekly requirement was replaced by a 96 hour Frequency in ITS SR 3.3.1.2. Because no specific requirement exists in CTS to perform this twice weekly calibration at equal intervals, testing could be performed with more than a 96 hour interval between them. This change to a 96 hour Frequency provides requirements consistent in format with NUREG-1430, while maintaining testing on a Frequency roughly equivalent to CTS requirements. In addition, ITS SR 3.3.1.2 requires calibration if the calorimetric heat balance is greater than 2% from the power range channel output. Although the CTS did not specifically state a criteria for calibration, ANO-1 station procedures do require channel calibration if a deviation of greater than 2% as described above is evident. Therefore, no new restriction than that already imposed at ANO-1 results from incorporation of ITS SR 3.3.1.2.

TECHNICAL CHANGE -- MORE RESTRICTIVE

M1 CTS Table 3.5.1-1 Reactor Protection System section Item 1 requirements for the manual push-button have been replaced with ITS 3.3.2.

ITS 3.3.2 has an Applicability of MODES 1 and 2, and MODES 3, 4, and 5 with any CONTROL ROD drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal. The equivalent CTS requirements were, by implication, applicable while above hot shutdown (MODE 3). This implied Applicability was based on the action requirements of Table 3.5.1-1 Note 1. The adoption of ITS 3.3.2 Applicability will specifically require this function to be OPERABLE in MODES 1 and 2. In addition, the RPS Manual Trip Function will be required to be OPERABLE while in MODES 3, 4 and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal. The adoption of this Applicability has been made to provide requirements consistent with those presented in NUREG-1430.

The adoption of ITS 3.3.2 ACTION A provides for a one hour restoration period prior to entry into Condition B or C. Adoption of this new ACTION provides a specific period of time for restoration, which did not exist in CTS. The one hour Completion Time of Required Action A.1, when added to the Completion Times of Required Actions B.1, B.2, and C.1, in each case, results in total times which are more restrictive than CTS requirements. ITS 3.3.2 ACTION A was adopted to provide requirements consistent with NUREG-1430.

Due to the adoption of an Applicability which includes MODES other than MODES 1 and 2, the adoption of ITS 3.3.2 ACTION C was required. This ACTION provides requirements appropriate for exiting the MODES of Applicability for this LCO. This ITS ACTION requirement, which is more restrictive than current requirements, was adopted to provide requirements consistent with NUREG-1430.

M2 Testing requirements on the anticipatory trip functions within the Reactor Protection System (RPS) have been changed to provide requirements consistent with the testing of other Functions within RPS. These changes have been made for both the turbine trip and the loss of main feedwater pump anticipatory reactor trips. These changes provide requirements consistent with those of NUREG-1430.

Specifically, the CHANNEL CHECK Frequency was changed from monthly to 12 hours. This 12 hour Frequency is consistent with the CHANNEL CHECK requirements, in CTS, for other RPS trip functions and is consistent with NUREG-1430. Additionally, the CHANNEL FUNCTIONAL TEST Frequency has been changed from "PC - Prior to going Critical if not done within previous 31 days" to 31 days. This change ensures that each of these channels is functionally tested each month, not just once prior to criticality. This monthly testing is consistent with the CTS testing requirements for other RPS trip functions and is consistent with the requirements of NUREG-1430.

M3 The adoption of ITS 3.3.1 ACTION D and ITS 3.3.2 ACTION B provides requirements more restrictive than those in CTS Table 3.5.1-1 Note 1. The adoption of these ACTIONS will require entry into MODE 3 within 6 hours as opposed to the current 12 hours. Additionally, these ACTIONS will require all CRD trip breakers to be opened within 6 hours. These changes are being made to improve consistency between the requirements of the ITS and NUREG-1430.

M4 Heat balance calibration requirements found in CTS Table 4.1-1 Item 3 have been replaced by ITS SR 3.3.1.2. This change includes the addition of criteria of when an adjustment of the power range instruments is required. This detail was not provided in CTS and therefore its addition represents more restrictive requirements. This change was made to provide testing requirements consistent with those found in NUREG-1430.

3.3.1-04

M5 The actions for inoperable Electronic (SCR) Trip Relays in CTS Table 3.5.1-1 Note 23 have been replaced by ITS 3.3.4 ACTION C. (Note: CTS term "Electronic (SCR) Trip Relay" is considered equivalent to ITS term "electronic trip assembly (ETA) relays.") CTS allows up to 48 hours to restore a single inoperable electronic (SCR) trip relay after which, the device is required to be in the trip (open) condition within the next hour. In the event more than one electronic trip relay, in a channel, is inoperable, all electronic (SCR) trip devices, in the channel, are to be tripped within one hour. These requirements have been replaced by ITS 3.3.4 ACTION C. This ACTION contains no provision for a 48 hour delay prior to requiring additional action to be taken. Because action to compensate for a single inoperable ETA is required sooner by ITS than by CTS, this change is more restrictive.

Additionally, because no actions were specified in CTS in the event the actions of Table 3.5.1-1 Note 23 were not completed, the addition of ITS 3.3.4 ACTION D is also more restrictive. With no additional actions specified, entry into CTS 3.0.3 would be appropriate, upon failure to comply with Table 3.5.1-1 Note 23. CTS 3.0.3 would allow up to 13 hours to reach hot shutdown (MODE 3) conditions. ITS 3.3.4 ACTION D requires the unit to be placed in MODE 3, with either the CRD trip breakers open, or power removed from the CRD system, within 6 hours.

ITS 3.3.4 ACTIONS C and D are being adopted to provide requirements consistent with NUREG-1430 and to provide ACTIONS for the ETA relays which are consistent with the ACTIONS required for the other CRD trip devices.

M6 ITS 3.3.4 Condition A has been adopted replacing CTS Table 3.5.1-1 Note 25. More restrictive requirements are represented by Condition A in that 48 hours were previously allowed for restoration of the inoperability followed by one hour in which to trip the breaker. Condition A will allow a total of 48 hours for restoration and either tripping (opening) the breaker or removing power from it. This reduction in total Completion Time from 49 hours to 48 hours is adopted to provide requirements consistent with NUREG-1430.

M7 CTS Table 3.5.1-1 Note 3 allowed continued operation above hot shutdown with the required source range instrument channel inoperable provided at least one intermediate range instrument was indicating greater than 1E-10 amps. No specific requirement existed in CTS to initiate repairs on this inoperable instrument. This CTS requirement has been replaced by ITS 3.3.9 ACTION B. The adoption of ACTION B will continue to allow operation above MODE 3 with the required source range instrument channel inoperable. However, the additional requirement to initiate action to repair the inoperable instrument channel within 1 hour is included. This additional requirement has been adopted to provide requirements consistent with NUREG-1430.

M8 CTS Table 3.5.1-1 RPS Functional Units 3 and 4, intermediate range instrument channels and source range instrument channels, both indicate that the actions of Note 1 are required in the event that the required instrument channel is inoperable. Note 1 requires that the unit be placed in hot shutdown (MODE 3) within 12 hours. No actions are specified in CTS to deal with their inoperability while in MODE 3 or below.

The requirements of Table 3.5.1-1 Note 1, as applied to the source range and intermediate range instrument channels, have been replaced by ITS 3.3.9 ACTION A and ITS 3.3.10 ACTION A, respectively. These new requirements are more restrictive in that they provide additional ACTIONS not required by CTS. These new ACTIONS provide requirements which ensure that the unit is placed in an acceptable condition to compensate for the inoperability of either the required source range instrument or the required intermediate range instrument. These additional ACTIONS are being adopted to provide requirements which are consistent with NUREG-1430 requirements.

M9 CHANNEL CALIBRATION requirements for the source range and intermediate range instruments on an 18 month Frequency have been adopted. Adoption of ITS SR 3.3.9.2 and ITS SR 3.3.10.3 represent more restrictive requirements because no equivalent requirements exist in CTS. These CHANNEL CALIBRATION requirements have been adopted to provide testing requirements consistent with NUREG-1430.

M10 The Required Action to be performed in the event the source range and/or intermediate range instruments are not operable is added to CTS Table 3.5.1-1. This action is illustrated in ITS 3.3.10 ACTION A. The adoption of this ITS ACTION presents more restrictive requirements in that unlimited continued operation with one or more monitors inoperable will no longer be allowed. ITS 3.3.10 ACTION A has been adopted to provide requirements consistent with NUREG-1430.

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M11 Requirements for the Reactor Protection System (RPS) Nuclear Overpower Low Setpoint function and the Shutdown Bypass RCS High Pressure function have been adopted in ITS Table 3.3.1-1. The adoption of these requirements includes the ACTION requirements of ITS 3.3.1 ACTION E. This ACTION has been added to provide an appropriate ACTION, where none existed in CTS, to deal with the inoperability of the shutdown bypass functions within the RPS. The adoption of ITS 3.3.1 ACTION E provides a Condition and Required Action consistent with NUREG-1430 and provides a Completion Time consistent with the Completion Times for other RPS ACTIONS in ITS 3.3.1. The Surveillance Requirements for these shutdown bypass functions were also adopted. These SRs were adopted to provide testing requirements consistent with those presented in NUREG-1430 for these functions.

M12 The "Applicable MODES or Other Specified Conditions" column of ITS Table 3.3.1-1 has been adopted to provide specific Applicability requirements for the individual RPS functions where no specific requirement existed in the ANO-1 CTS. Each of the RPS functions in CTS Table 3.5.1-1 was by implication required to be OPERABLE while above hot shutdown (MODE 3). This implied Applicability was based on the action requirements of Table 3.5.1-1 Note 1. The addition of specific Applicability requirements while in other than MODES 1 and 2, where none existed previously, is consistent with NUREG 1430.

M13 CTS 3.5.1.3 is revised to reflect ITS 3.3.1 ACTIONS A and B. The requirements are essentially equivalent except that no Completion Time is included in the CTS. This addition of a specific Completion Time is considered more restrictive because no specific requirements similar to these exist in CTS. These changes have been made to provide requirements consistent with NUREG-1430.

M14 ITS 3.3.3 Reactor Protection System (RPS)--Reactor Trip Module (RTM) LCO, Applicability, and ACTIONS have been adopted. ANO-1 CTS requirements for the RTMs were presented only as testing requirements in Table 4.1-1 Item 1, Protective Channel Coincidence Logic. The adoption of these ITS requirements represent more restrictive requirements because no specific requirements similar to these exist in CTS. These changes have been made to provide requirements consistent with NUREG-1430.

M15 ITS 3.3.4, CONTROL ROD Drive (CRD) Trip Devices, Applicability, and ACTIONS D and E, and their respective Bases, have been adopted. Each of these items was adopted to provide requirements consistent with NUREG-1430. The adoption of these items represents more restrictive requirements because no specific requirements similar to these exist in CTS.

M16 ITS 3.3.9, Source Range Neutron Flux, Applicability has been adopted. While the MODE 2 Applicability was implied by the action requirements of Table 3.5.1-1 Note 1 and 2, no specific statement of Applicability for these instrument channels exists in CTS. Although at least one source range instrument channel is now maintained OPERABLE while operating in MODES 2, 3 , 4, and 5, the adoption of the Applicability statement in ITS 3.3.9 represents more restrictive requirements in that specific requirements will now exist where none existed previously. Appropriate ACTION requirements were also adopted in the form of ITS 3.3.9 ACTIONS A. These additional requirements have been adopted to provide requirements consistent with NUREG-1430 while maintaining the CTS requirement to require only one source range instrument channel while in these MODES.

M17 ITS 3.3.10, Intermediate Range Neutron Flux, Applicability has been adopted. While the MODE 2 Applicability was implied by the action requirements of Table 3.5.1-1 Note 1 and 2, no specific statement of Applicability for these instrument channels exists in CTS. Although at least one intermediate range instrument channel is now maintained OPERABLE while operating in MODE 2, the adoption of the Applicability statement in ITS 3.3.10 represents more restrictive requirements in that specific requirements will now exist where none existed previously. Appropriate ACTION requirements were also adopted in the form of ITS 3.3.10 ACTION A. These additional requirements have been adopted to provide requirements consistent with NUREG-1430 while maintaining the CTS requirement to require only one intermediate range instrument channel while in these MODES.

M18 ITS 3.3.1 Condition C has been adopted. This adoption provides for specific Required Actions in the event ITS 3.3.1 Required Action and Completion Times of Condition A or B are not met. The proposed Condition C also provides Required Actions in the event that more than 2 channels are inoperable.

ITS 3.3.1 ACTIONS A and B are essentially the requirements of CTS 3.5.1.3 (with the exception of Completion Times: see DOC M13). Failure to comply with CTS 3.5.1.3 would have required entry into CTS 3.0.3 which would have allowed continued operation above hot shutdown (MODE 3) for up to 13 hours. The adoption of ITS 3.3.1 Condition C will in many cases result in requirements to enter MODE 3 within 6 hours. These more restrictive requirements are being adopted to provide requirements consistent with NUREG-1430 requirements.

When applied to the specific condition of more than 2 channels inoperable, the adoption of ITS 3.3.1 Condition C is administrative in nature. The adoption of Condition C requires entry into specific ACTION requirements. Similarly, CTS Table 3.5.1-1 column 5 specifies that the actions of CTS Table 3.5.1-1 Note 1 apply when less than 2 channels are OPERABLE (more than 2 channels inoperable). The changes in Completion Times related to CTS Table 3.5.1-1 Note 1 are discussed in other Discussions of Changes.

M19 The Allowable Values for ITS Table 3.3.1 Function 9 Main Turbine Trip (Control Oil Pressure) and Function 10 Loss of Main Feedwater Pumps (Control Oil Pressure) have been added. Because these values were not specified in CTS their adoption in ITS represents additional restrictions These additions has been made to provide requirements consistent with NUREG-1430 requirements.

TECHNICAL CHANGE -- LESS RESTRICTIVE

L1 The requirements of CTS 3.5.1.6 have been replaced by those of ITS 3.3.4 Condition B. The requirements of ITS 3.3.4 Condition B are less restrictive than those found in CTS 3.5.1.6 and are consistent with the actions required by CTS Table 3.5.1-1 Note 24.

Adoption of the requirements of ITS 3.3.4 Condition B allows a time period of up to one hour to either trip or remove power from an inoperable CONTROL ROD Drive (CRD) trip breaker as opposed to the 30 minutes allowed by CTS 3.5.1.6 for removing power from a trip device failed in the untripped state. The change from 30 minutes to one hour is consistent with the requirements of CTS Table 3.5.1-1 Note 24 for an inoperable CRD trip breaker (not specifically failed in the untripped state), and is consistent with NUREG-1430 requirements. The change from 30 minutes to one hour removes the confusion provided by having different times specified in CTS for completing identical actions for very similar failures. The additional 30 minutes only slightly increases the time the unit is allowed to operate in this condition and therefore, the likelihood of an additional failure occurring during this time which could prevent the CONTROL RODS from inserting, is very slight.

Additionally, ITS 3.3.4 will allow unrestricted operation for an unlimited period of time with an inoperable CRD trip breaker, even if the breaker is in the untripped state, provided power is removed from that breaker. CTS 3.5.1.6 requires that the untripped breaker be repaired and the remaining trip devices be tested within 8 hours of detection or the unit be placed in Hot Shutdown (MODE 3) within the next 4 hours. Operation for an indefinite period of time, with inoperable untripped CRD trip devices, from which power has been removed, is acceptable because removal of power from the untripped device accomplishes the same result as tripping the device. Because the function of the of the CRD trip device has been accomplished, continued operation is acceptable. Removal of the requirement to test the remaining trip devices when one is found failed in the untripped state is also appropriate. Failure of one of these devices in the untripped state in no way increases the likelihood of the failure of another of these devices. Further, any indication of a common mode failure results (through administrative controls) in timely evaluation of the condition and appropriate determination of the need for additional testing.

L2 Heat balance calibration requirements found in CTS Table 4.1-1 Item 3 have been replaced by ITS SR 3.3.1.2. The Frequency for performance of heat balance calibrations of twice weekly was changed to a 96 hour frequency in ITS SR 3.3.1.2. Although no specific requirement exists in CTS to perform this twice-weekly calibration at equal intervals, if this twice weekly requirement is interpreted as being required on equal intervals, the change to 96 hours represents a less restrictive change. Based on this interpretation of current requirements, the 96 hour Frequency would provide an additional 12 hours between required performances of this calibration. This additional time between performances of heat balance calibrations is acceptable, based on operating experience, due to the very slow divergence between power range instruments and heat balance power level during steady state operation. This change to a 96 hour Frequency provides requirements consistent in format with NUREG-1430, while maintaining testing on a Frequency which approximates CTS requirements.

L3 The Applicability of the requirements to have OPERABLE RPS anticipatory trip functions on a loss of main feedwater and on a turbine trip has been changed from the CTS Applicability of greater than 5% reactor power for both, to ≥ 10% RTP for Loss of Main Feedwater Pumps and ≥ 45% RTP for Main Turbine Trip. ANO-1 CTS allows the trip on loss of main feedwater function to be bypassed up to 10% reactor power and the trip on turbine trip function to be bypassed up to 45% reactor power. The adoption of the ITS Applicability will maintain requirements consistent with CTS in that the trip function for Loss Main Feedwater will be required to be in service (not bypassed) and OPERABLE prior to exceeding 10% RTP. Similarly, the trip function for Main Turbine Trip will be required to be in service (not bypassed) and OPERABLE prior to exceeding 45% RTP. Reference to the bypass permissives and removal capability is relocated to the applicable Bases since the bypass functions are not required for channel operability. The ITS requirements are less restrictive than the CTS requirements in that the trip on Loss of Main Feedwater Pumps will no longer be required to be OPERABLE between 5% and 10% power and the trip on Main Turbine Trip will no longer be required to be OPERABLE between 5% and 45% power.

3.3.1-02

This change in Applicability is acceptable because it provides OPERABILITY requirements for the anticipatory trip functions which are consistent with their safety function. By requiring these trip functions to be OPERABLE while operating at reactor power levels at which they were allowed to be bypassed, CTS provided OPERABILITY requirements which were inconsistent with their safety function.

Additionally, the action requirements provided by CTS 3.5.1.9.3 have been replaced by ITS 3.3.1 Conditions F and G. This change provides for ACTION requirements which are consistent with the Applicability of ITS 3.3.1 for these functions and which are consistent with NUREG-1430. This change will provide requirements to exit the MODES of Applicability in the event these functions are INOPERABLE. Removal of the requirements to go to hot shutdown (MODE 3), which is well below even the CTS MODES of Applicability, is consistent with the philosophy presented in NUREG-1430. This philosophy is to require that either the equipment be restored or the unit exit the MODES of Applicability.

L4 Not Used.

3.3.1-03 L5 Not Used

L6 Not Used.

L7 ITS 3.3.4 Required Actions and Completion Times C.1, C.2 and C.4 have been adopted as alternatives to the requirements presented in CTS Table 3.5.1-1 Note 23. Required Action C.1 specifically allows for a CONTROL ROD group with an inoperable ETA relay to be placed on a power supply which has OPERABLE or open ETA relays. Required Action C.2 allows for a safety rod group which is being powered from the auxiliary power supply to be returned to its DC hold power supply if one or both ETA relays associated with the auxiliary power supply become inoperable. Required Action C.4 provides an additional alternative for dealing with inoperable ETA relays. This Required Action allows the corresponding AC breaker to be opened to compensate for the loss of the ETA relay. These allowances all provide new flexibility which was not previously specified in CTS. Required Actions C.1 and C.2 are each acceptable alternatives to opening an inoperable ETA relay because they each place the affected CONTROL ROD group on a power supply which will ensure that the rods are de-energized upon a reactor trip. Required Action C.4 is an acceptable alternative because it interrupts power to the affected CONTROL ROD group in a manner equivalent to opening the ETA relay.

ITS 3.3.4 ACTION C provides similar requirements for each inoperable ETA relay regardless of the total number of inoperable relays. CTS Table 3.5.1-1 Note 23 requires that with two or more inoperable relays, all relays associated with the channel, whether OPERABLE or inoperable, be opened within one hour. This requirement places unnecessary restrictions on the operation of the unit, by requiring OPERABLE equipment to be removed from service. The adoption of ITS 3.3.4 ACTION C is less restrictive in that it will not require any action to be taken to open a specific ETA relay if that relay is not inoperable. Removal of the requirement to open all ETA relays in a channel, in the event two or more ETA relays are inoperable, is acceptable because the Required Actions and Completion Times of ITS 3.3.4 Condition C provide requirements to compensate for the loss of function for each ETA relay individually within the same relatively short 1 hour time frame.

L8 CTS Table 3.5.1-1 Note 25 requires that any Control Rod Drive Trip Breaker with an inoperable undervoltage or shunt trip function be restored within 48 hours or the trip breaker must be opened. Adoption of ITS 3.3.4 Required Action A.2 will allow power to be removed from the inoperable trip breaker as an alternative to opening the breaker. This additional allowance is less restrictive in that it provides additional flexibility in dealing with trip breakers with an inoperable diverse trip function. The allowance for removing power from a trip breaker as an alternative to opening the breaker is currently allowed by CTS Table 3.5.1-1 Note 24. However, this note is not specifically applicable to the inoperability of the diverse trip function for the trip breakers. The addition of ITS 3.3.4 Required Action A.2 provides consistent ACTION requirements to compensate for inoperable CRD trip breakers whether or not the inoperability is due to failure of a diverse trip function. The Completion Times in ITS will remain, as they are in CTS, significantly different for a CRD trip breaker with an inoperable diverse trip function, as opposed to one which is inoperable for any other reason.

L9 The CTS requirement to check and calibrate the power range instruments against the incore instruments monthly was found in CTS Table 4.1-1 Item 4. The calibration was also required to be performed within some unspecified period of time after each startup if not performed within the previous week. These CTS requirements have been replaced with ITS SR 3.3.1.3 and modified by TSTF 342, Rev. 1. The TSTF recognized that power range outputs needed to be compared with and adjusted to incore instrumentation, which would not have necessarily been performed under the definition of CHANNEL CALIBRATION. The CHANNEL CALIBRATION is already required by ITS SR 3.3.1.6 with an appropriate frequency of 18 months. The comparison and channel adjustments approved under the TSTF is currently the practice at ANO-1 to meet the CTS requirements. The adoption of the ITS SR and the specified 31 day Frequency, including the changes incorporated through TSTF 342, Rev. 1, represent less restrictive requirements in that the adjustment will no longer be required following each startup if not performed within the previous week. Removal of the required adjustment, within a week prior to or following each startup, is acceptable because deviation between the AXIAL POWER IMBALANCE indicated by the power range instruments and that indicated by the incore instruments generally occurs slowly. Adoption of the 31-day Frequency and TSTF 342, Rev. 1 maintains requirements consistent with NUREG-1430.

3.3.1-05

L10 CTS Table 3.5.1-1 Note 2 indirectly provided a statement of Applicability for the OPERABILITY requirements for the source range and intermediate range instrument channels. This note provides a relaxation of action requirements when "2 of 4 power range instrument channels are greater than 10% rated power." The Applicability of ITS 3.3.9 and 3.3.10 does not require either the source range instrument channel or the intermediate range instrument channel to be maintained OPERABLE above MODE 2. This represents a relaxation of requirements, by removing the requirement to take actions in the event that either the required source range instrument channel or the required intermediate range instrument channel is inoperable, when above 5% RTP (ITS) but less than or equal to 10% rated power, as indicated on the power range instruments (CTS).

3.3.9-02
3.3.10-02

Startup rate (SUR) information is provided by the source range monitors (SRM) and the intermediate range monitors (IRM). High SUR conditions will prohibit control rod withdrawal. However, the SRMs reach saturation about midway between the point of criticality and the point at which power range monitors (PRM) come on scale (less than 1% power). Once saturated, the SRM SUR function that is designed to support an approach to criticality, becomes ineffective in supporting further increases in reactor power. The CTS applicability originated from the design of the SRMs that existed at initial startup. These instruments have previously been replaced by gamma metric-type monitors. To avoid unnecessary costs, the existing interlock relay used to defeat the SUR interlock at 10% power was maintained, even though the SRM SUR function is invalid in the power range. Although the IRM SUR function may still support limiting rapid power increases due to CEA withdrawal up to the time when the interlock is bypassed at 10% power, the SRM and IRM SUR rod withdrawal prohibits are not credited in any accident analyses. Furthermore, a wide range gamma metric monitor is available to track reactor power from well below the point of criticality to full power. Once the Point of Adding Heat (POAH) is reached shortly after achieving criticality, other power indicators become available such as reactor coolant temperature, steam bypass valve, and auxiliary feed pump response. Since the SUR interlock function is not relied upon in the safety analyses and is not needed in the power range, other plant systems can be used as an indication of plant power, the SRMs are saturated before reaching the power range, and the wide range gamma metrics monitor provides additional power indication through the IRM range, the NUREG-1430 Applicability for SRMs and IRMs is acceptable.

This change also allows for less than a full decade of overlap between the PRMs and the IRMs, but continues to require IRMs until the PRMs are on-scale. This change is being made to provide clear statements of Applicability for these specifications which are consistent with the requirements of NUREG-1430.

L11 Not used.

L12 The Note modifying ITS 3.3.1.2 has been adopted. No specific allowance is provided in the ANO-1 CTS which removes the requirement to perform this calibration while in MODE 1 at low power levels. Adoption of this Note provides an exception to the performance of this calibration which recognizes the difficulty in its performance and the limitations of the calorimetric while operating at very low power levels. Below 20% RTP ANO-1 calculates heat balance power level based totally upon the primary system parameters. Above 20% RTP, the secondary system parameters are also considered since they are generally more accurate at higher power levels. By allowing the delay in performance of this calibration until RTP is above 20%, a generally more accurate calorimetric (one including secondary system parameters) is available. The Note does not imply, however, that the function of the instrumentation should not be met. The required 18-month calibration of the channels provides assurance that the channel is functional and will respond to power changes. In addition, although below 20% RTP the comparison of the power instrumentation to a calorimetric or incore system is not appropriate, the status of the instrumentation is none-the-less tracked. Station procedures require operators to use heat balance power to assess rod index limits. Operators also use other indications to verify the approximate accuracy of power instrumentation below 20% RTP by comparing with known power indicators such as steam bypass valve position, auxiliary feed pump output, turbine load, etc. Delaying SR 3.3.1.3 performance until > 20% RTP does not prevent operators from assessing the availability of power instrumentation and, therefore, is acceptable. Furthermore, the adoption of this Note does not involve no significant hazards considerations. This allowance is being adopted to provide requirements consistent with NUREG-1430.

3.3.1-06

L13 The Note (2) modifying ITS SR 3.3.1.3 has been adopted. This Note allows a delay in performance of this SR until the unit is above 20% RTP. This allowance is appropriate due to the usable range of the incore nuclear instruments which are required for the performance of this SR. Below about 20% the incore nuclear instruments are not capable of providing reliable accurate indication of AXIAL POWER IMBALANCE. Adoption of this Note provides a specific relaxation of requirements where none existed in CTS. The Note does not imply, however, that the function of the instrumentation should not be met. The required 18-month calibration of the channels provides assurance that the channel is functional and will respond to power changes. In addition, although below 20% RTP the comparison of the power instrumentation to a calorimetric or incore system is not appropriate, the status of the instrumentation is none-the-less tracked. Station procedures require operators to use heat balance power to assess rod index limits. Operators also use other indications to verify the approximate accuracy of power instrumentation below 20% RTP by comparing with known power indicators such as steam bypass valve position, auxiliary feed pump output, turbine load, etc. Delaying SR 3.3.1.3 performance until > 20% RTP does not prevent operators from assessing the availability of power instrumentation and, therefore, is acceptable. Furthermore, the adoption of this Note does not involve no significant hazards considerations. This adoption is being made to provide requirements consistent with NUREG-1430.

3.3.1-06

CTS DISCUSSION OF CHANGES

L14 The CTS Table 4.1-1 Item 5 requirement, to perform CHANNEL FUNCTIONAL TEST on the intermediate range instrument channel, prior to each startup, if not performed within the previous week, has not been retained in the ITS. The requirement to perform this CHANNEL FUNCTIONAL TEST on a 31 day Frequency, was, however, retained as ITS 3.3.10.2. With the deletion of the required testing within 7 days of start-up, this testing will simply be required each 31 days. This 31 day frequency will also require the performance of the CHANNEL FUNCTIONAL TEST within 31 days of a start-up. This extension of the Frequency of this test from 7 days to 31 days prior to a start-up is acceptable based on operating experience which indicates that the intermediate range instrumentation is highly reliable and is not likely to experience an undetected failure during the extended period between tests.

L15 The requirement to perform a CHANNEL FUNCTIONAL TEST on the source range instrument channel within 7 days prior to start-up has not been retained in the ITS. This requirement was located in CTS 4.1-1 Item 6. This deletion has been made to provide testing requirements, for the required source range instrument channel, consistent with NUREG-1430.

A new requirement, to perform a CHANNEL CALIBRATION on the required source range instrument channel, on an 18 month Frequency, has been adopted in the ITS. No similar CHANNEL CALIBRATION requirements, for the source range instruments, existed in CTS. Because this calibration, by definition, encompasses the CHANNEL FUNCTIONAL TEST, performance of this calibration will ensure that testing, consistent with CTS requirements, continues to be required. The Frequency of this testing will, however, now be based strictly on the time since its last performance and not dependent upon whether or not the unit is in start-up. This change is acceptable, based on operating experience, which indicates that the source range instrumentation is highly reliable, and is no more susceptible to undetected failures within 7 days of start-up, than at any other time that the instrumentation is required to be OPERABLE.

The addition of the requirement to perform the CHANNEL CALIBRATION is discussed elsewhere in these Discussions of Change.

L16 NUREG 1430 LCO 3.3.10 Condition A requirements were added to CTS Table 3.5.1-1 (See DOC M10). These requirements are revised to allow minor positive reactivity additions that are a result of plant temperature changes when no intermediate or source range neutron flux monitor is operable. During such conditions, various unit operations must continue, including the control of RCS temperature. The addition of this allowance is acceptable since reactivity controls are maintained for the plant mode in which the condition exists. This change is consistent with NUREG-1430, as modified by generic change TSTF-286, Revision 2.

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LESS RESTRICTIVE -- ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 This information has been moved to the Bases or TRM. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe an acceptable method of compliance or detail of unit design. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Bases will be controlled by the Bases Control Process in Chapter 5 of the proposed Technical Specifications. The details of performance of the surveillances have generally been relocated to the TRM along with various other requirements not appropriate for inclusion in the ITS. Changes to the TRM will be controlled by 10 CFR 50.59. This change is consistent with NUREG-1430.

CTS Location	New Location
1.5.6	Bases 3.3.1, SR 3.3.1.2
Table 2.3-1 Note (a)	Bases 3.3.1, BACKGROUND
Table 2.3-1 Note (b)	Bases 3.3.1, ASA
Table 2.3-1 Note (b)	Bases 3.3.1, LCO
Table 2.3-1 Note (b)	Bases 3.3.1, APPLICABILITY
Table 2.3-1 Note (c)	Bases 3.3.1, ASA
Table 2.3-1 Note (c)	Bases 3.3.1, LCO
Table 2.3-1 Note (c)	Bases 3.3.1, APPLICABILITY
Table 2.3-1 Note (d)	Bases 3.3.1, ASA
Table 2.3-1 Note (d)	Bases 3.3.1, LCO
Table 2.3-1 Note (d)	Bases 3.3.1, APPLICABILITY
3.5.1.3	Bases 3.3.1, BACKGROUND
3.5.1.5	Bases SR 3.3.1.1, SR 3.3.9.1, SR 3.3.10.1
Table 3.5.1-1 Note 7	Bases 3.3.9, BACKGROUND
Table 3.5.1-1 Note 7	Bases 3.3.10, BACKGROUND
Table 3.5.1-1 Note 23	Bases 3.3.4, LCO
Table 3.5.1-1 Note 26	Bases 3.3.4, BACKGROUND
Table 3.5.1-1 Columns 1 and 2	Bases 3.3.1, 3.3.2, 3.3.3, 3.3.4, 3.3.9, and 3.3.10, BACKGROUND
Table 4.1-1 Item 2 Note (1)	Bases 3.3.4, SR 3.3.4.1

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3.3.1

1.5.4 Instrument Channel Calibration

An instrument channel calibration is a test, and adjustment (if necessary), to establish that the channel output responds with acceptable range and accuracy to known values of the parameter which the channel measures or an accurate simulation of these values. Calibration shall encompass the entire channel, including equipment actuation, alarm or trip and shall be deemed to include the channel test.

<LATER> (1.0)

1.5.5 Heat Balance Check

LATER

A heat balance check is a comparison of the indicated neutron power and core thermal power.

1.5.6 Heat Balance Calibration

An adjustment of the power range channel amplifiers output to agree with the core thermal power as determined by a weighted primary and secondary heat balance considering all heat losses. ~~Between 0 and 15% power, only the primary heat balance is considered. From 15 to 100% power the heat balance is weighted linearly with only the secondary heat balance being considered at 100% power.~~

LA1 Bases

1.6 POWER DISTRIBUTION

1.6.1 Quadrant Power Tilt

Quadrant power tilt shall be defined by the following equation and is expressed as a percentage

<LATER> (1.0)

$$100\left(\frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}} - 1\right)$$

LATER

1.6.2 Reactor Power Imbalance

Reactor power imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

A1

Applicability

Applies to instruments monitoring reactor power, reactor power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.

Objective

To provide automatic protection action to prevent any combination of process variables from exceeding a safety limit.

Specification

LCO 3.3.1

2.3.1 The reactor protection system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3-1 and the Protection System Maximum Allowable Setpoint for Axial Power Imbalance as given in the COLR.

LA1 Bases

Bases

A2

The reactor protection system consists of four instrument channels to monitor each of several selected plant conditions which will cause a reactor trip if any one of these conditions deviates from a pre-selected operating range to the degree that a safety limit may be reached.

The trip setting limits for protection system instrumentation are listed in Table 2.3-1. The safety analysis has been based upon these protection system instrumentation trip setpoints plus calibration and instrumentation errors.

Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 104.9 percent of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is the value used in the safety analysis.

A. Overpower Trip Based on Flow and Imbalance

The power level trip set point produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant-flow accident from high power. Analysis has demonstrated that the specified power-to-flow ratio is adequate to prevent a DNBR of less than 1.30 (BAW-2) or 1.18 (BWC) should a low flow condition exist due to any electrical malfunction.

A2

The power level trip setpoint produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate.

The flux/flow ratios account for the maximum calibration and instrumentation errors and the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

No penalty in reactor coolant flow through the core was taken for an open core vent valve because of the core vent valve surveillance program during each refueling outage. For safety analysis calculations the maximum calibration and instrumentation errors for the power level were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The reactor power imbalance (power in top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio so that the boundaries of the Protection System Maximum Allowable Setpoints for Axial Power Imbalance in the COLR are produced. The power-to-flow ratio reduces the power level trip associated reactor power-to-reactor power imbalance boundaries by the value specified in the COLR for a 1 percent flow reduction.

B. Pump Monitors

In conjunction with the power imbalance/flow trip, the pump monitors prevent the minimum core DNBR from decreasing below 1.30 (BAW-2) or 1.18 (BWC) by tripping the reactor due to the loss of reactor coolant

12

pumps(s). The pump monitors also restrict the power level for the number of pumps in operation.

C. RCS Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Table 2.3-1 for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient.(2)

The low pressure (1800 psig) and variable low pressure (COLR) trip setpoint shown in Table 2.3-1 have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction.(2,3)

To account for the calibration and instrumentation errors, the accident analysis used the protective limit specified in the COLR.

D. Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618F) shown in Table 2.3-1 has been established to prevent excessive core coolant temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip setpoint of 620 F.

E. Reactor Building Pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

F. Shutdown Bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. A nuclear overpower trip set point of $\leq$5.0 percent of rated power is automatically imposed during reactor shutdown.

2. A high reactor coolant system pressure trip setpoint of 1720 psig is automatically imposed.

3.3.1

A2

The purpose of the 1720 psig high-pressure trip setpoint is to prevent normal operation with part of the reactor protection system bypassed. This high-pressure trip setpoint is lower than the normal low-pressure trip setpoint so that the reactor must be tripped before the bypass is initiated. The overpower trip setpoint of ≤5.0 prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation (5) would be available to remove 5.0 percent of rated power if none of the reactor coolant pumps were operating.

<u>REFERENCES</u>

(1) FSAR, Section 14.1.2.3

(2) FSAR, Section 14.1.2.2

(3) FSAR, Section 14.1.2.7

(4) FSAR, Section 14.1.2.8

(5) FSAR, Section 14.1.2.6

A1

Table 2.3-1
Reactor Protection System Trip Setting Limits

Table 3.3.1-1 Allowable Values

Function #		Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power, 75%)	One Reactor Coolant Pump Operating in Each Loop (Nominal Operating Power, 49%)[d]	Shutdown Bypass
1.a/1.b	Nuclear power, % of rated, max	104.9	104.9	104.9	5.0[a]
8	Nuclear Power based on flow[b] and imbalance, % of rated, max	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Bypassed
7	Nuclear Power based on pump monitors, % of rated, max[c]	NA	NA	55	Bypassed
3/11	High RC system pressure, psig, max	2355	2355	2355	1720[a]
4	Low RC system pressure, psig. min	1800	1800	1800	Bypassed
5	Variable low RC system pressure, psig, min	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Bypassed
2	RC temp, F, max	618	618	618	618
6	High reactor building pressure, ~~psig~~, max	4 (18.7 psia)	4 (18.7 psia)	4 (18.7 psia)	4 (18.7 psia)

<LATER (3.4A)> LATER A1 A12

(a) Automatically set when other segments of the RPS (as specified) are bypassed.
(b) Reactor coolant system flow, %
(c) The pump monitors also produce a trip on (a) loss of two RC pumps in one RC loop, and (b) loss of one or two RC pumps during two-pump operation.

LA1 Bases

(d) Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hrs. with the reactor critical.

<LATER (3.4A)> LATER

<Add Table 3.3.1-1 Allowable Values for Functions 9 & 10> M19

<Add Table 3.3.1-1 Note (a)> A1

3.3.1

3.3.1
3.3.4
3.3.10

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operational Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

(A1)

Objectives

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

3.5.1.1 Startup and operation are not permitted unless the requirements Table 3.5.1-1, columns 3 and 4 are met.

(A3) & LATER

& (LATER) (3.3B, 3.3C, 3.3D, 3.4B)

3.5.1.2 In the event the number of protection channels operable falls be the limit given under Table 3.5.1-1, Columns 3 and 4, operation shall be limited as specified in Column 5.

3.5.1.3 For on-line testing or in the event of a protection instrument or channel failure, a key operated channel bypass switch associated with each reactor protection channel may be used to lock the channel trip relay in the untripped state as indicated by a light.

(LA1) BASES

3.3.1 RA B.1

Only one channel shall be locked in the untripped state or contain inoperable functions in the untripped state at any one time. In the event more than one protection channel contains inoperable functions in the untripped state, or a protection channel or function becomes inoperable concurrent with another protection channel locked in the untripped state, within 1 hour implement the actions required by Table 3.5.1-1 Note 6.

(A3)

Only one channel bypass key shall be accessible for use in the control room.

(LA1) BASES

3.3.1 RA A.1 & A.2
3.3.1 RA B.2.1 & B.2.2

While operating with an inoperable function unbypassed in the untripped state, the remaining RPS key operated channel bypass switches shall be tagged to prevent their operation.

(M13)

3.5.1.4 The key operated shutdown bypass switch associated with each reactor protection channel shall not be used during reactor power operation except during channel testing.

(R) TRM

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Bases SR 3.3.1.1, SR 3.3.9.1, SR 3.3.10.1

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall be maintai in the source range until the one decade overlap is achieved.

(LA1)

3.3.4 RA B.2

3.5.1.6 In the event that one of the trip devices in either of the sour supplying power to the control rod drive mechanisms fails in the untripped state, the power supplied to the rod drive mechanisms through the failed trip device shall be manually removed within 30 (1 hr) minutes following detection. The condition will be corrected and the remaining trip devices shall be tested within eight hours following detection. If the condition is not corrected and the remaining trip devices are not tested within the eight-hour period the reactor shall be placed in the hot shutdown condition within an additional four hours.

(L1)

⟨LATER⟩ (3.4B)

3.5.1.7 The Decay Heat Removal System isolation valve closure setpoints shall be equal to or less than 340 psig for one valve and equal to or less than 400 psig for the second valve in the suction line. The relief valve setting for the DHR system shall be equal to or less than 450 psig. — LATER

⟨LATER⟩ (3.3 D)

3.5.1.8 The degraded voltage monitoring relay settings shall be as follows: — LATER

a. The 4.16 KV emergency bus undervoltage relay setpoints shall be >3115 VAC but <3177 VAC.

b. The 460 V emergency bus undervoltage relay setpoints shall be >423 VAC but <431 VAC with a time delay setpoint of 8 seconds ± 1 second.

3.5.1.9 ~~The following Reactor Trip circuitry shall be operable as indicated:~~ — A1

Table 3.3.1-1, Function 10, "Applicable MODES"

1. Reactor trip upon loss of Main Feedwater shall be operable ~~(as determined by Specification 4.1.a and item 35 of Table 4.1-1)~~ at greater than ~~5% reactor power.~~ ~~(May be bypassed up to)~~ 10% reactor power.) — A1 — L3

Table 3.3.1-1, Function 9 "Applicable MODES"

2. Reactor trip upon Turbine Trip shall be operable ~~(as determined by Specification 4.1.a and item 41 of Table 4.1-1)~~ at greater than ~~5% reactor power.~~ ~~(May be bypassed up to)~~ 45% reactor power.) — A1 — L3

Table 3.3.1-1, Cond. ref. from RA C.1 Function 10 - 3.3.1 RA G.1 Function 9 - 3.3.1 RA F.1

3. If the requirements of Specifications 3.5.1.9.1 or 3.5.1.9.2 cannot be met, ~~restore the inoperable trip within 12 hours or bring the plant to a hot shutdown condition.~~ Cannot be met, reduce power to ≤ 10% RTP, Reduce power to ≤ 45% RTP — L3

3.5.1.10 ~~Deleted~~ — A1

⟨LATER⟩ (3.3C)

3.5.1.11 For on-line testing of the Emergency Feedwater Initiation and Control (EFIC) system channels during power operation only one channel shall be locked into "maintenance bypass" at any one time. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed. — LATER

⟨LATER⟩ (3.3 D)

3.5.1.12 The Containment High Range Radiation Monitoring Instrumentation shall be operable with a minimum measurement range from 1 to 10^7 R/hr. — LATER

3.3.1
3.3.4
3.3.10

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless the requirements of Table 3.5.1-1, Columns 3 and 4, are met.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column 4 (Table 3.5.1-1). This is in agreement with redundancy and single failure criteria of IEEE 279 as described in FSAR, Section 7.

There are four reactor protection channels. Normal trip logic is two-out-of-four. Required trip logic for the power range instrumentation channels is two-out-of-three. Minimum trip logic on other instrumentation channels is one-out-of-two.

The four reactor protection channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided with alarm and lights to indicate when that channel is bypassed. There will be one reactor protection system channel bypass switch key permitted in the control room. Upon the discovery of inoperable functions in any one reactor protection channel, the effect of the failure on the reactor protection syst and other interconnected systems is evaluated. The affected reactor protect channel may be placed in channel bypass, remain in operation in a degraded condition, or placed in the tripped condition as determined by operating conditions and management judgment. This action allows placing the plant in the safest condition possible considering the extent of the failure, plant conditions, and guidance from plant management. Should the failure in the reactor protection channel prohibit the proper operation of another system, the appropriate actions for the affected system are implemented. Administra controls are established to preclude placing a reactor protection channel ir channel bypass when any other reactor protection channel contains an inopera function in the untripped state. A2

Each reactor protection channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used. R TRM

The source range and intermediate range nuclear flux instrumentation scales overlap by one decade. This decade overlap will be achieved at 10^{-10} amps c the intermediate range scale.

The ESAS employs three independent and identical analog channels, which supply trip signals to two independent, identical digital subsystems. In order to actuate the safeguards systems, two out of three analog channels must trip. This will cause both digital subsystems to trip. Tripping of either digital subsystem will actuate all safeguards systems associated with that digital subsystem. A2

Because only one digital subsystem is necessary to actuate the safeguards systems and these systems are capable of tripping even when they are being tested, a single failure in a digital subsystem cannot prevent protective action.

3.3.1
3.3.4
3.3.10

~~Removal of a module required for protection from a RPS channel will cause that channel to trip, unless that channel has been bypassed, so that only one channel of the other three must trip to cause a reactor trip. Thus, sufficient redundancy has been built into the system to cover this situation.~~

~~Removal of a module required for protective action from an analog ESAS channel will cause that channel to trip, so that only one of the other two must trip to actuate the safeguards systems. Removal of a module required for protective action from a digital ESAS subsystem will not cause that subsystem to trip. The fact that a module has been removed will be continuously annunciated to the operator. The redundant digital subsystem is still sufficient to indicate complete ESAS action.~~

~~The testing schemes of the RPS, the ESAS, and the EFIC enables complete system testing while the reactor is operating. Each channel is capable of being tested independently so that operation of individual channels may be evaluated.~~

~~The EFIC is designed to allow testing during power operation. One channel may be placed in key locked "maintenance bypass" prior to testing. This will bypass only one channel of EFW initiate logic. An interlock feature prevents bypassing more than one channel at a time. In addition, since the EFIC receives signals from the NI/RPS, the maintenance bypass from the NI/RPS is interlocked with the EFIC. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed. Prior to placing a channel of EFIC in maintenance bypass, any NI/RPS channel containing inoperable functions in the untripped state is evaluated for its effect on EFIC. Only the EFIC channel corresponding to the NI/RPS channel containing the inoperable function may be placed in maintenance bypass unles it can be shown that the failure in the NI/RPS channel has no effect on EFIC actuation, actions are taken to ensure EFIC actuation when required, or the appropriate actions of Table 3.5.1-1 are implemented. The EFIC can be teste from its input terminals to the actuated device controllers. A test of the EFIC trip logic will actuate one of two relays in the controllers. Activati of both relays is required in order to actuate the controllers. The two relays are tested individually to prevent automatic actuation of the compone The EFIC trip logic is two (one-out-of-two).~~ (A2)

~~Reactor trips on loss of all main feedwater and on turbine trips will sense the start of a loss of OTSG heat sink and actuate earlier than other trip signals. This early actuation will provide a lower peak RC pressure during the initial over pressurization following a loss of feedwater or turbine trip event. The LOFW trip may be bypassed up to 10% to allow sufficient margin for bringing the MFW pumps into use at approximately 7%. The Turbine Trip may be bypassed up to 45% based on BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," October 1985 and the NRC Safety Evaluation Report for BAW-1893 issued from Mr. D. M. Crutchfield to Mr. J. H. Taylor via letter dated April 25, 1986.~~

~~The Automatic Closure and Isolation System (ACI) is designed to close the Decay Heat Removal System (DHRS) return line isolation valves when the Reactor Coolant System (RCS) pressure exceeds a selected fraction of the DHRS design pressure or when core flooding system isolation valves are opened. The ACI is designed to permit manual operation of the DHRS return line isolation valves when permissive conditions exist. In addition, the ACI is designed to disallow manual operation of the valves when permissive conditions do not exist.~~

3.3.1
3.3.4
3.3.10

~~Power is normally supplied to the control rod drive mechanisms from two separate parallel 480 volt sources. Redundant trip devices are employed in each of these sources. If any one of these trip devices fails in the untripped state, on-line repairs to the failed device, when practical, will be made and the remaining trip devices will be tested. Four hours is ample time to test the remaining trip devices and, in many cases, make on-line repairs.~~

~~The Degraded Voltage Monitoring relay settings are based on the short term starting voltage protection as well as long term running voltage protection. The 4.16 KV undervoltage relay setpoints are based on the allowable starting voltage plus maximum system voltage drops to the motor terminals, which allows approximately 78% of motor rated voltage at the motor terminals. The 460V undervoltage relay setpoint is based on long term motor voltage requirements plus the maximum feeder voltage drop allowance resulting in a 92% setting of motor rated voltage.~~

A2

~~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 recommendation of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power 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."~~

~~The subcooled margin monitors (SMM), and core-exit thermocouples (CET), Reactor Vessel Level Monitoring System (RVLMS) and Hot Leg Level Measurement System (HLLMS) are a result of the Inadequate Core Cooling (ICC) instrumentation required by Item II.F.2 NUREG-0737. The function of the ICC instrumentation is to increase the ability of the plant operators to diagnose the approach to and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37 and are not required by the accident analysis, nor to bring the plant to cold shutdown conditions. The Reactor Vessel Level Monitor is provided as a means of indicating level in the reactor vessel during accident conditions. The channel operability of the RVLMS is defined as a minimum of three sensors in the upper plenum region and two sensors in the dome region operable. When Reactor Coolant Pumps are running, all except the dome sensors are interlocked to read "invalid" due to flow induced variables that may offset the sensor outputs. The channel operability of the HLLMS is defined as a minimum of one wide range and any two of the narrow range transmitters in the same channel operable. If the equipment is inaccessible due to health and industrial safety concerns (for example, high radiation area, low oxygen content of the containment atmosphere) or due to physical location of the fault (for example, probe failure in the reactor vessel), then operation may continue until the next scheduled refueling outage and a report filed.~~

3.3.1
3.3.4
3.3.10

The principal function of the Control Room Isolation-High Radiation is to provide an enclosed environment from which the unit can be operated following an uncontrolled release of radioactivity. Due to the unique arrangement of the shared control room envelope, one control room isolation channel receives a high radiation signal from the ANO-1 control room ventilation intake duct monitor and the redundant channel receives a high radiation signal from the ANO-2 control room ventilation intake duct monitor. With no channel of the control room radiation monitoring system operable, the CREVS must be placed in a condition that does not require the isolation to occur (i.e., one operable train of CREVS is placed in the emergency recirculation mode of operation). Reactor operation may continue indefinitely in this state.

To support loss of main feedwater analyses, steam line/feedwater line break analyses, SBLOCA analyses, and NUREG-0737 requirements, the EFIC system is designed to automatically initiate EFW when:

1. all four RC pumps are tripped
2. both main feedwater pumps are tripped
3. the level of either steam generator is low
4. either steam generator pressure is low
5. ESAS ECCS actuation (high RB pressure or low RCS pressure)

The EFIC system is also designed to isolate the affected steam generator on a steam line/feedwater line break and supply EFW to the intact generator according to the following logic: A2

- If both SG's are above 600 psig, supply EFW to both SG's.
- If one SG is below 600 psig, supply EFW to the other SG.
- If both SG's are below 600 psig, but the pressure difference between the two SG's exceeds 100 psig, supply EFW only to the SG with the higher pressure.
- If both SG's are below 600 psig and the pressure difference is less than 100 psig, supply EFW to both SG's.

At cold shutdown conditions all EFIC initiate and isolate functions are bypassed except low steam generator level initiate. The bypassed functions will be automatically reset at the values or plant conditions identified in Specification 3.5.1.15. "Loss of 4 RC pumps" initiate and "low steam generator pressure" initiate are the only shutdown bypasses to be manually initiated during cooldown. If reset is not done manually, they will automatically reset. Main feedwater pump trip bypass is automatically removed above 10% power.

REFERENCE

FSAR, Section 7.1

<Add 3.3.1 Appl.> A1

<Add 3.3.1 Condition C> M18

<Add 3.3.1 Condition E> M11

<Add 3.3.1 Surveillance Requirements - NOTE> A1

LA1 BASES

A3

A7

3.3.1 LCO — **Table 3.5.1-1 Instrumentation Limiting Conditions for Operation (Note 6)**

REACTOR PROTECTION SYSTEM

RAI 3.3.1-03

A6

	Functional Unit	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met (A3)
3.3.2 LCO	1. Manual pushbutton	1	1	1	0	Note 1 (M1)
T3.3.1-1, #1a	2. ~~Power range~~ nuclear overpower instrument channel	4	2	~~3 (Note 4)~~ 2	~~1 (Note 4)~~ 0	Note 1 (A17)
3.3.10 LCO	3. Intermediate range instrument channels	2	Note 7	1	0	Notes ~~1,~~ 2 (A1)
3.3.9 LCO	4. Source range instrument channels	2	Note 7	1	0	Notes ~~1,~~ 2, 3 (M8)
T3.3.1-1, #2	5. Reactor coolant temperature instrument channels	4	2	2	1	Note 1
T3.3.1-1, #5 &	6. Pressure-temperature instrument channels	4	2	2	1	Note 1
T3.3.1-1, #8	7. Flux/imbalance/flow instrument channels	4	2	2	1	Note 1
	8. Reactor coolant pressure					
T3.3.1-1, #3	a. High reactor coolant pressure instrument channels	4	2	2	1	Note 1 (A1)
T3.3.1-1, #4	b. Low reactor coolant pressure instrument channels	4	2	2	1	Note 1
T3.3.1-1, #7	9. Power/number of pumps instrument channels	4	2	2	1	Note 1
T3.3.1-1, #6	10. High reactor building pressure channels	4	2	2	1	Note 1

<Add Table 3.3.1-1 APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS for Functions 1 thru 8 & 11, including T3.3.1-1 Notes a, b, c & d.> M12

3.3.1
3.3.2
3.3.9
3.3.10

⟨Add Table 3.3.1-1 Function 1b, Nuclear Overpower Low Setpoint, Condition & SRs⟩ — M11

⟨Add Table 3.3.1-1 Function 11, Shutdown Bypass RCS High Pressure, Condition & SRs⟩ — LA1 BASES

Table 3.5.1-1 (Cont'd)

A3 A7

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REACTOR PROTECTION SYSTEM (Cont'd)

Functional Unit	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met
T3.3.1-1, function 10 — 11. Reactor trip upon loss of Main Feedwater	4	2	2	1	Notes 1, 15
function 9 — 12. Reactor trip upon turbine trip	4	2	2	1	Notes 1, 16
3.3.4 LCO c — 13. Electronic (SCR) Trip Relay	2	2	2	0	Note 23
14. Control Rod Drive Trip Breakers					
3.3.4 LCO a — A. AC Breakers	2	2	2	0	Notes 24, 25
3.3.4 LCO b — B. DC Breakers ~~(Note 26)~~	2	2	2	0	Notes 24, 25

A3 A1 A8 A1

44a

LA1 BASES

⟨Add 3.3.2 Appl, Conditions A & C⟩ — M1

⟨Add 3.3.3 LCO, Appl, and ACTIONS⟩ — M14

⟨Add 3.3.4 ACTIONS - NOTE⟩ — A9

⟨Add 3.3.4 Appl, Conditions D & E⟩ — M15

⟨Add 3.3.9 Appl, Condition A⟩ — M16

3.3.1 3.3.2 3.3.3 3.3.4 3.3.9

A3

MODE 3 with the CRD breakers open within 6 hrs.

TABLE 3.5.1-1 (Cont'd)

Notes:

3.3.1 Cond. D
3.3.2 Cond. B & <LATER>(3.3B/C/D & 3.4B)

1. Initiate a shutdown using normal operating instructions and place the reactor in the hot shutdown condition within 12 hours if the requirements of Columns 3 and 4 are not met. — M3 & LATER

3.3.9 Appl.
3.3.10 Appl.

2. When 2 of 4 power range instrument channels are greater than ~~10%~~ 5% rated power, hot shutdown is not required. — L10

3.3.9 Cond. B

3. When 1 of 2 intermediate range instrument channels is greater than 10-10 amps, ~~hot shutdown is not required.~~ initiate action to restore source range channel within 1 hr. — M7

4. ~~For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours, after which Note 1 applies.~~ — A17

<LATER> (3.3B, 3.4B)

5. ~~If the requirements of Columns 3 or 4 cannot be met within an additional 48 hours, place the reactor in the cold shutdown condition within 24 hours.~~ — LATER

3.3.1 RA B.1 & <LATER> (3.3B & 3.3C)

6. The minimum number of operable channels may be reduced to 2, provided that the system is reduced to 1 out of 2 coincidence by tripping the remaining channel. Otherwise, the actions required by Column 5 shall apply. — A1 & LATER

7. ~~These channels initiate control rod withdrawal inhibits not reactor trips at -10% rated power. Above 10% rated power, those inhibits are bypassed.~~ — LA1 BASES

<LATER> (3.3B)

8. ~~If any one component of a digital subsystem is inoperable, the entire digital subsystem is considered inoperable. Hence, the associated safety features are inoperable and Specification 3.3 applies.~~ — LATER

<LATER> (3.3D)

9. ~~The minimum number of operable channels may be reduced to one and the minimum degree of redundancy to zero for a maximum of 24 hours, after which Note 1 applies.~~

10. ~~With the number of operable channels less than required, either restore the inoperable channel to operable status within 30 days, or be in hot shutdown within 12 hours.~~

11. ~~With the number of operable channels less than required, isolate the electromatic relief valve within 4 hours, otherwise Note 9 applies.~~ — LATER

<Added LCO 3.3.10 Condition A> — M10

Plant temperature changes are allowed provided the temperature changes are accounted for in the SDM calculations.

RAI 3.3.1-03

ANO-332

ANO-333

3.3.1
3.3.2
3.3.9
3.3.10

(A2)

TABLE 3.5.1-1 (Cont'd)

<LATER> (3.3D)

12. With the number of operable channels less than required, either return the indicator to operable status within 24 hours, or verify the block valve closed and power removed within an additional 24 hours. If the block valve cannot be verified closed within the additional 24 hours, de-energize the electromatic relief valve power supply within the following 12 hours.

13. Channels may be bypassed for not greater than 30 seconds during reactor coolant pump starts. If the automatic bypass circuit or its alarm circuit is inoperable, the undervoltage protection shall be restored within 1 hour, otherwise, Note 14 applies.

LATER

<LATER> (3.3D & 3.8)

14. With the number of channels less than required, restore the inoperable channels to operable status within 72 hours or be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

LATER

Table 3.3.1-1 Function 10 Appl Function 9 Appl & <LATER> (3.3C)

15. This trip function may be bypassed at up to 10% reactor power.

16. This trip function may be bypassed at up to 45% reactor power.

L3 & LATER

<LATER> (3.3D)

17. With no channel operable, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

18. With one channel inoperable, restore the inoperable channel to operable status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

LATER

<LATER> (3.3C)

19. This function may be bypassed below 750 psig OTSG pressure. Bypass is automatically removed when pressure exceeds 750 psig.

LATER

<LATER> (3.3D)

20. With one channel inoperable, (1) either restore the inoperable channel to operable status within 7 days, or (2) prepare and submit a Special Report to the Commission pursuant to Specification 6.12.5 within 30 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status. With both channels inoperable, initiate alternate methods of monitoring the containment radiation level within 72 hours in addition to the actions described above.

21. With one channel inoperable, restore the inoperable channel to operable status within 30 days or be in hot shutdown within 72 hours unless containment entry is required. If containment entry is required, the inoperable channel must be restored by the next refueling outage. If both channels are inoperable, restore the inoperable channels within 30 days or be in HOT SHUTDOWN within 12 hours.

LATER

⟨Add 3.3.4 RA C.1, C.2, & C.4⟩ A3 M5 L7

Table 3.5.1-1 (cont'd)

3.3.4 RA C.3

23. ~~With the number of operable Electronic (SCR) Trip relays one less than the total number of Electronic (SCR) Trip relays in a channel, restore the inoperable Electronic (SCR) Trip relay to operable status in 48 hours or place the SCRs associated with the inoperable Electronic (SCR) Trip relay in trip in the next hour. With two or more Electronic (SCR) Trip relays inoperable, place all Electronic (SCR) Trip relays associated with that channel in trip in the next hour. This requirement does not apply to the Electronic Trip channels associated with Group 8 Regulating Power Supply.~~ LA1 Bases

3.3.4 RA B.1

3.3.4 RA B.2

24. With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied: A1

a. Within 1 hour:

1. Place the inoperable channel in the tripped condition, or

2. Remove power supplied to the control rod trip device associated with the inoperable channel.

b. ~~One additional channel may be bypassed for up to 4 hours for surveillance testing and the (in) operable channel above may be bypassed for up to 30 minutes in any 24-hour period when necessary to test the trip breaker associated with the logic of the channel being tested. The inoperable channel above shall not be bypassed to test the logic of a channel of the trip system associated with the inoperable channel.~~ [Sic] A13

3.3.4 Cond. A 25. With one of the Control Rod Drive Trip Breaker diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status ~~in 48 hours~~ or place the breaker in trip ~~in the next hour.~~ M6

or remove power from the trip breaker L8

26. ~~Interrupts motor power to the Safety Groups of control rods only.~~ LA1 Bases

27. ~~Deleted~~ A1

3.3.4

3.3.1
3.3.2
3.3.3
3.3.4
3.3.9
3.3.10

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications: LATER

<LATER> (5.0)

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requiremetns of any Technical Specification.

4.1 OPERATIONAL SAFETY ITEMS

A1

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

a. The minimum frequency and type of surveillance required for reactor protective system and engineered safeguards system instrumentation when the reactor is critical shall be as stated in Table 4.1-1. A3 & R TRM

3.3.1
3.3.2
3.3.3
3.3.4
3.3.9
3.3.10

OPERATIONAL SAFETY ITEMS (continued) A1

4.1 (Continued)

& R TRM

A3

& <LATER> (3.3B, 3.3C, 3.3D)

b. Equipment and sampling test shall be performed as detailed in Table 4.1-2 and 4.1-3.

c. Discrepancies noted during surveillance testing will be corrected and recorded.

<LATER> (3.2)

d. A power distribution map shall be made to verify the expected power distribution at periodic intervals at least every 10 effective full power days using the incore instrumentation detector system. LATER

A2

BASES

4.0.1 through 4.0.5 Establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR 50.36(c)(3):

"Surveillance Requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

4.0.1 Establishes the requirement that surveillances must be performed during the operational modes or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a mode or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.

3.3.1
3.3.2
3.3.3
3.3.4
3.3.9
3.3.10

BASES (continued)

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an operational mode or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, that is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable. A2

4.1 Bases

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation. A2 & R TRM

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels shall be calibrated at least twice weekly (during steady state operating conditions) against a heat balance standard to compensate for instrumentation drift. During nonsteady state operation, the nuclear flux channels shall be calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters. A2

3.3.1
3.3.2
3.3.3
3.3.4
3.3.9
3.3.10

Other channels are subject only to "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed once every 18 months.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies for the nuclear flux (power range) channels, and once every 18 months for the process system channels is considered acceptable.

Testing

On-line testing of reactor protective channel and EFIC channels is required once every 4 weeks on a rotational or staggered basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel.

All reactor protective channels will be tested before startup if the individual channel rotational frequency has been discontinued or if outage activities could potentially have affected the operability of one or more channels. A rotation will then be established to test the first Channel one week after startup, the second Channel two weeks after startup, the third Channel three weeks after startup, and the fourth Channel four weeks after startup. A2

The established reactor protective system instrumentation and EFIC test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action, all instrumentation associated with the protective channels will be tested after which the rotational test cycle is started again. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protective channels coincidence logic and control rod drive trip breakers are trip tested every quarter. The trip test checks all logic combinations and is to be performed on a rotational basis. The logic and breakers of the four protective channels shall be trip tested prior to startup and their individual channels trip tested on a cyclic basis. Discovery of a failure requires the testing of all channel logic and breakers, after which the trip test cycle is started again.

3.3.1
3.3.2
3.3.3
3.3.4
3.3.9
3.3.10

The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the status of the equipment and systems to assure safe operation. A2

& P
TRM

REFERENCE

FSAR Section 7.1.2.3.4

<Add SR 3.3.1.2 - NOTE> L12

<Add SR 3.3.1.3 - NOTE> L13

<Add SR 3.3.1.6 - NOTE> A10

A3

RAI 3.3.1-04

RAI 3.3.1-06

Table 4.1-1
Instrument Surveillance Requirements

A1 LA1 Bases

	Channel Description	Check	Test	Calibrate	Remarks
SR 3.3.3.1	1. ~~Protective Channel Coincidence Logic~~ Reactor Trip Module	NA	Q	NA	
SR 3.3.4.1	2. Control Rod Drive Trip Breaker	NA	Q(1)	NA	~~(1) To include independent testing of the shunt and undervoltage trip attachments.~~ A1
SR 3.3.1.2	3. Power Range Amplifier	NA	NA	~~T/W(1)~~ Adjust Power range Channel output if Calorimetric exceeds power range channel by ≥ 2% RTP	A18 M4 (1) Heat balance calibration ~~twice~~ every 96 hours ~~weekly under steady state operating conditions,~~ ~~daily~~ 24 hours under non-steady state operating conditions. L2 A5
SR 3.3.1.3 SR 3.3.1.1 SR 3.3.1.3 Table 3.3.1-1 Function 1a + Function 8	4. ~~Power Range Channel~~ Nuclear Overpower and Nuclear Overpower RCS Flow and measured AXIAL POWER IMBALANCE	S M(1)	~~M~~	M(1)(2)	(1) Using core instrumentation. (2) Axial offset upper and lower chambers ~~monthly and after each startup if not done previous week.~~ every 31 days L9
SR 3.3.10.1 SR 3.3.10.2	5. Intermediate Range Channel	S	~~P~~ M	~~NA~~ R	A11
SR 3.3.9.1	6. Source Range Channel	S(1)	~~P~~	~~NA~~ R	A6 ~~(1) When in service.~~ A1
Table 3.3.1-1 Function 2	7. Reactor Coolant Temperature Channel	S	M	R	M9 Add SR 3.3.9.2 and SR 3.3.10.3 with NOTES
Function 3	8. High Reactor Coolant Pressure Channel	S	M	R	L14 L15
Function 4	9. Low Reactor Coolant Pressure Channel	S	M	R	A1
Function 8	10. Flux-Reactor Coolant Flow Comparator	S	~~M~~	R	A11
Function 5	11. Reactor Coolant Pressure Temperature Comparator	S	M	R	
Function 7	12. Pump Flux Comparator	S	M	R	
		SR 3.3.1.1	SR 3.3.1.4	SR 3.3.1.6	

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3.3.1
3.3.3
3.3.4
3.3.9
3.3.10

(A3)

Table 4.1-1 (cont.)

	Channel Description	Check	Test	Calibrate	Remarks	
Table 3.3.1-1 Function 6	13. High Reactor Building Pressure Channel	S SR 3.3.1.1	M SR 3.3.1.4	R SR 3.3.1.6		(A1)
	14. High Pressure Injection Logic Channel	NA	M	NA		
	15. High Pressure Injection Analog Channels					
<LATER> (3.3B, 3.3D)	a. Reactor Coolant Pressure Channel	S	M(1)	R	(1) Including test of shutdown bypass function (ECCS bypass function).	
	b. Reactor Building 4 psig Channel	S	M	R		
<LATER> (3.3B)	16. Low Pressure Injection Logic Channel	NA	M	NA		
	17. Low Pressure Injection Analog Channels					
<LATER> (3.3B, 3.3D)	a. Reactor Coolant Pressure Channel	S	M(1)	R	(1) Including test of shutdown bypass function (ECCS bypass function).	LATER
	b. Reactor Building 4 psig Channel	S	M	R		
	18. Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	M	NA		
	19. Reactor Building Emergency Cooling and Isolation System Analog Channels					
	a. Reactor Building 4 psig Channels	S	M	R		

70

A3

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks	
<LATER> (3.3D)	~~29. High and Low Pressure Injection Systems: Flow Channels~~	~~NA~~	~~NA~~	~~R~~		LATER
<LATER> (3.4B)	~~30. Decay heat removal system isolation valve automatic closure and interlock system~~	~~S(1)(2)~~	~~M(1)(3)~~	~~R~~	~~(1) Includes RCS Pressure Analog Channel (2) Includes CFT Isolation Valve Position (3) At least once every refueling shutdown, with Reactor Coolant System Pressure greater than or equal to 200 psig, but less than 300 psig, verify automatic isolation of the decay heat removal system from the Reactor Coolant System on high Reactor Coolant System pressure.~~	LATER
	~~31. Deleted~~					A1
<LATER> (3.8)	~~32. Diesel generator protective relaying starting interlocks and circuitry~~	~~M~~	~~Q~~	~~NA~~		LATER
<LATER> (3.8)	~~33. Off-site power undervoltage and protective relaying interlocks and circuitry~~	~~W~~	~~R(1)~~	~~R(1)~~	~~(1) Shall be tested during refueling shutdown to demonstrate selective load shedding interlocks function during manual or automatic transfer of Unit 1 auxiliary load to Startup Transformer No. 2.~~	LATER
<LATER> (3.3D)	~~34. Borated water storage tank level indicator~~	~~W~~	~~NA~~	~~R~~		LATER
Table 3.3.1-1 Function 10	35. Reactor trip upon loss of main feedwater circuitry	~~M~~ 12 hour SR 3.3.1.1	~~PC~~ 31 days SR 3.3.1.4	R SR 3.3.1.6		M2

3.3.1

A3

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks	
<LATER> (3.5)	~~36. Boric Acid Addition Tank~~					
	~~a. Level Channel~~	~~NA~~	~~NA~~	~~R~~		LATER
	~~b. Temperature Channel~~	~~M~~	~~NA~~	~~R~~		LATER
<LATER> (3.30)	~~37. Degraded Voltage Monitoring~~	~~W~~	~~R~~	~~R~~		R
	~~38. Sodium Hydroxide Tank Level Indicator~~	~~NA~~	~~NA~~	~~R~~		TRM
<LATER> (3.2)	~~39. Incore Neutron Detectors~~	~~M(1)~~	~~NA~~	~~NA~~	~~(1) Check Functioning~~	LATER
<LATER> (3.30)	~~40. Emergency Plant Radiation Instruments~~	~~M(1)~~	~~NA~~	~~R~~	~~(1) Battery Check~~	LATER
Table 3.3.1-1 Function 9	41. Reactor Trip Upon Turbine Trip Circuitry	~~M~~ 12 hours SR 3.3.1.1	~~PC~~ 31 days SR 3.3.1.4	R SR 3.3.1.6		M 2
	~~42. Deleted~~					A1

3.3.1

(A3)

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks
<LATER> (3.3B)	43. ESAS Manual Trip Functions				LATER
	a. Switches & Logic	NA	R	NA	
	b. Logic	NA	M	NA	
SR 3.3.2.1	44. Reactor Manual Trip	NA	P	NA	(A1)
<LATER> (3.4B)	45. Reactor Building Sump Level	NA	NA	R	LATER
<LATER> (3.3D)	46. EFW Flow Indication	M	NA	R	LATER

3.3.2

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(A3)

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
d. SG A High Range Level High-high	S	M	R	
e. SG B High Range Level High-high	S	M	R	
57. Containment High Range Radiation Monitors	D	M	R	
58. Containment Pressure-High	M	NA	R	
59. Containment Water Level-Wide Range	M	NA	R	
60. Low Temperature Overpressure Protection Alarm Logic	NA	R	R	
61. Core-exit Thermocouples	M	NA	R	
62 Electronic (SCR) Trip Relays	NA	Q	NA	
63 RVLMS	M	NA	R	
64 HLLMS	M	NA	R	

<LATER> (3.3c)

LATER

<LATER> (3.3D)

LATER

<LATER> (3.4B)

LATER

<LATER> (3.3D)

LATER

SR 3.3.4.1

(A1)

<LATER> (3.3D)

LATER

NOTE:

S - Each Shift
W - Weekly
M - Monthly
D - Daily

T/W - Twice per Week
Q - Quarterly
P - Prior to each startup if not done previous week
B/M - Every 2 months

R - Once every 18 months
PC - Prior to going Critical if not done within previous 31 days
NA - Not Applicable
SA - SA Twice per Year

<LATER> (3.3B 3.3C 3.3D 3.4B)

(A4) + LATER + (R) TRM

3.3.4

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"R" - Relocation of requirements:

Relocating requirements which do not meet the Technical Specification selection criteria to documents with an established control program allows the Technical Specifications to be reserved only for those conditions or limitations upon reactor operation which are necessary to adequately limit the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, thereby focusing the scope of Technical Specifications.

Therefore, requirements which do not meet the Technical Specification selection criteria in 10 CFR 50.36 have been relocated to other controlled license basis documents. This regulation addresses the scope and purpose of Technical Specifications. In doing so, it establishes a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. These criteria are as follows:

Criterion 1: Installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier.

Criterion 4: A structure, system or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The application of these criteria is provided in the "Application of Selection Criteria to the ANO-1 Technical Specifications." Requirements which met the criteria have been included in the proposed improved Technical Specifications. Entergy Operations proposes to remove the requirements which do not meet the criteria from the Technical Specifications and relocate the requirements to a suitable owner controlled document. The requirements in the relocated Specifications will not be affected by this Technical Specification change. Entergy Operations will initially continue to perform the required operation and maintenance to assure that the requirements are satisfied. Relocating specific requirements for systems or variables will have no impact on the system's operability or the variable's maintenance, as applicable.

License basis document control mechanisms, such as 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls," will be utilized for the relocated Specifications as they will be placed in other controlled license basis documents. This would allow Entergy Operations to make changes to these requirements, without NRC approval, as allowed by the applicable regulatory requirements. These controls are considered adequate for assuring structures, systems and components in the relocated Specifications are maintained operable and variables in the relocated Specifications are maintained within limits.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the ANO-1 Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled license basis document and maintained pursuant to the applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the affected requirement will be relocated to an owner controlled license basis document for which future changes will be evaluated pursuant to the requirements of the applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

"A" - Administrative changes to requirements:

Reformatting and rewording the remaining requirements in accordance with the style of the improved Babcock & Wilcox Standard Technical Specifications in NUREG-1430 will make the Technical Specifications more readily understandable to plant operators and other users. Application of the format and style will also assure consistency is achieved between specifications. As a result, the reformatting and rewording of the Technical Specifications has been performed to make them more readily understandable by plant operators and other users. During this reformatting and rewording process, no technical changes (either actual or interpretational) to the Technical Specifications were made unless they were identified and justified.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change involves reformatting and rewording of the existing Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, there is no technical change to the requirements and therefore, there is no significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"LA" - Less restrictive, Administrative deletion of requirements:

Portions of some Specifications provide information that is descriptive in nature regarding the equipment, system(s), actions or surveillances. This information is proposed to be deleted from the specification and relocated to other license basis documents which are under licensee control. These documents include the TS Bases, Safety Analysis Report (SAR), Technical Requirements Manual, and Programs and Manuals identified in ITS Section 5, "Administrative Controls." The removal of descriptive information is permissible, because the documents containing the relocated information will be controlled through the applicable process provided by the regulatory requirements, e.g., 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls." This will not impact the actual requirements but may provide some flexibility in how the requirement is conducted. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relocates requirements from the Technical Specifications to other license basis documents which are under licensee control. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the Technical Specifications to other license basis documents, which are under licensee control, are the same as the existing Technical Specifications. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"M" - More restrictive changes to requirements:

The ANO-1 Technical Specifications are proposed to be modified in some areas to impose more stringent requirements than previously identified. These more restrictive modifications are being imposed to be consistent with the improved Babcock & Wilcox Standard Technical Specifications. Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

The modification of the ANO-1 Technical Specifications and the changes made to achieve consistency within the specifications have been performed in a manner such that the most stringent requirements are imposed, except in cases which are individually evaluated.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change provides more stringent requirements for the ANO-1 Technical Specifications. These more stringent requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not impact the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated for ANO-1.

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements prevents a reduction in the margin of plant safety by:

a) Increasing the analytical or safety limit,
b) Increasing the scope of the specification to include additional plant equipment,
c) Increasing the applicability of the specification,
d) Providing additional actions,
e) Decreasing restoration times,
f) Imposing new surveillances, or
g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ITS Section 3.3A: Instrumentation - RPS

Entergy Operations has evaluated these proposed Technical Specification changes and has determined that they involve no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92(c) as indicated below:

3.3A L1

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

This change provides an extension of the Completion Time, for removing power from an inoperable control rod drive (CRD) trip breaker which is failed in the untripped position, from 30 minutes to 1 hour. This short extension of the Completion Time for a Required Action does not result in any hardware changes. The Completion Time for performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed) and the proposed Completion Time extension is short (and therefore limits the impact on probability). Also, an extension of the Completion Time provides additional opportunity to restore compliance with the requirements and avoid the increased potential for a transient during the shutdown process. Further, the Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because a change in the Completion Time does not change the assumed response of the equipment in performing its specified mitigation functions from that considered during the original Completion Time.

This change will also allow operation to continue indefinitely, with one CRD trip breaker failed in the untripped state, provided power is removed from the breaker. Once power has been removed from the inoperable breaker, the position of the breaker, tripped or untripped, is irrelevant, because the safety function of the breaker, to interrupt power to the CRDs, has been accomplished. Therefore, as long as the Required Action (removing power from the untripped inoperable CRD trip breaker) is accomplished, this change does not involve any increase in the probability or consequences of any accident previously evaluated.

The final portion of this change is to remove the requirement to test each of the other CRD trip breakers, in the event one of them is determined to be inoperable, in the untripped state. This change removes an unnecessary, additional performance of a surveillance which has been performed within its normally required Frequency. Not performing the surveillance would not affect any equipment which is assumed to be an initiator of any analyzed event. Further, since the surveillance continues to be performed on its normal Frequency, there is no impact on the capability of the system to perform its required safety function. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

3.3A L1 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Additionally, the proposed change will still ensure proper surveillances are required for all equipment considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Completion Time has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, the short extension of the Completion Time interval does not involve a significant reduction in the margin of safety.

The ability to promptly initiate the insertion of the CONTROL RODS continues to be provided, while one CRD trip breaker is in the untripped state with its power removed. Therefore, this change does not involve a significant reduction in a margin to safety.

The normal surveillance Frequency, for CRD trip breaker testing, has been shown, based on operating experience, to be adequate for assuring the equipment is available and capable of performing its intended function. Further, any indication of a common mode failure results (through administrative controls) in timely evaluation of the condition and appropriate determination of the need for additional testing. Additionally, the requirements of SR 3.0.4 (CTS 4.0.4) provide assurance the equipment is OPERABLE prior to beginning the functions for which it is required. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.3A L2

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change replaces the twice per week Frequency, for performance of a heat balance calibration of the power range instruments, with a 96 hour Frequency. This change allows less frequent performances of this Surveillance Requirement. A less frequent performance of a Surveillance Requirement does not result in any hardware changes. The Frequency of performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed) and since the proposed Frequency has been determined to be adequate to demonstrate reliable operation of the equipment. Further, the Frequency of performance of a surveillance does not significantly increase the consequences of an accident because, a change in Frequency does not change the assumed response of the equipment in performing its specified mitigation functions from that considered with the original Frequency. The power range instrument channels are used to support mitigation of the consequences of an accident; however, operating experience has shown 96 hours is sufficient to detect deviations between heat balance power and indicated reactor power. Further, this information is readily available to the operator, i.e., heat balance power indication and power range neutron flux indicators, to identify abnormalities. Therefore, this change does not involve an increase in the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for all equipment considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Changes in the surveilled parameter have been determined to be relatively slow during the proposed intervals, and the proposed Frequency has been determined to be sufficient to identify significant impact on compliance with the assumed conditions of the safety analysis. In addition, other indications continue to be available to indicate potential noncompliance. Therefore, an extended surveillance interval does not involve a significant reduction in the margin of safety.

3.3A L3

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The Applicability for the Loss of Main Feedwater Pumps reactor trip Function and the Main Turbine Trip reactor trip Function has been changed from 5% reactor power to ≥10% RTP and ≥45% RTP respectively. Similarly, the Required Actions have been revised to require only that the MODE of Applicability be exited. This change in Applicability and Required Actions for these functions does not result in any hardware changes. This change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed). Also, the changes do not change the assumed response of the equipment in performing its specified mitigation functions from that considered during the original Applicability since these trip functions were allowed by CTS to be bypassed during the Conditions which will be omitted from the revised Applicability. Therefore, the changes do not significantly increase the consequences of an accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper availability for the required anticipatory trip functions. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The Loss of Main Feedwater Pumps reactor trip Function and the Main Turbine Trip reactor trip Function provide anticipatory trips under certain operating conditions. In the conditions to be excluded from the Applicability, the trip functions are bypassed and provide no input to the safety analysis. Therefore, the changes do not involve a significant reduction in the margin of safety.

3.3A L4 - Not used.

3.3.1-03 **3.3A L5** – Not Used

3.3A L6 - Not used.

3.3A L7

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

This change provides Required Actions which allow OPERABLE equipment to remain in service to be available to perform its safety function rather than remove it from service due to inoperability of another portion of the channel. This change does not result in any hardware changes. This change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the RPS and CRD trip devices does not change (and therefore any initiation scenarios are not changed), and appropriate response of the RPS and CRD trip devices continues to be provided by the alternative Required Actions. Also, the change does not change the assumed response of the equipment in performing its specified mitigation functions from that considered in the safety analysis. Therefore, the changes do not significantly increase the consequences of an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper availability for the required trip devices. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The continued availability of OPERABLE trip devices is enhanced by the proposed change. Therefore, the change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.3A L8

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change provides Required Actions which allow alternative, equivalent compensatory activities for inoperable equipment to maintain the overall availability of the RPS safety function. This change does not result in any hardware changes. This change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the RPS does not change (and therefore any initiation scenarios are not changed), and appropriate response of the RPS continues to be provided by the alternative Required Actions. Also, the change does not change the assumed response of the equipment in performing its specified mitigation functions from that considered in the safety analysis. Therefore, the changes do not significantly increase the consequences of an accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper availability for the required trip devices. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The continued availability of the RPS trip devices is maintained by the proposed change. Therefore, the change does not involve a significant reduction in the margin of safety.

3.3A L9

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change does not result in any hardware changes or changes in operating methods. The change removes an unnecessary additional performance of a surveillance which has been performed within its normal monthly Frequency. Not performing the surveillance at the startup would not affect any equipment which is assumed to be an initiator of any analyzed event. Further, since the surveillance continues to be performed on its normal Frequency, there is no impact on the capability of the system to perform its required safety function. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure adequate surveillance is performed to identify any degradation of the power range instrumentation channel. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The margin of safety for a power range instrument channel is based on availability and capability of the instrument to perform its safety function. Since the monthly Frequency has been determined (through experience with CTS) to be adequate to confirm the availability and capability, the removal of an additional confirmatory check of the instrumentation does not impact that availability and capability. Therefore, this change does not involve a significant reduction in a margin of safety.

3.3A L10

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

The Applicability for the source range and intermediate range instrument channels of RPS is limited such that they are not required above MODE 2. Similarly, the Required Actions have been revised such that no actions are required if these channels are inoperable in MODE 1. This change in Applicability and Required Actions for these functions does not result in any hardware changes. This change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed). Also, the changes do not change the assumed response of the equipment in performing its specified mitigation functions from that considered in the safety analysis since the power range monitors provide on-scale indication prior to entering MODE 1. Therefore, the changes do not significantly increase the consequences of an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure appropriate availability for the instrument channels considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The instrument channels provide neutron power indication and control rod withdrawal inhibit interlocks (based on high startup rate), under low power operating conditions. In the conditions to be excluded from the Applicability, indication of neutron power is provided by the power range instrumentation channels. Additionally, the control rod withdrawal inhibit interlock functions are not credited by the safety analysis. Therefore, the changes do not involve a significant reduction in the margin of safety.

3.3A L11 - Not used.

3.3A L12

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

A Note is included which allows deferring the calorimetric heat balance for adjustment of the power range instrument channels of the RPS while at low power levels. This Note recognizes the limitations of a calorimetric at low power levels. This change in applicability for this Surveillance does not result in any hardware changes. The power range monitors are not considered as initiators for any previously analyzed accidents. As such, the change does not significantly increase the probability of occurrence of any analyzed event. Performance of this Surveillance at low power levels generally provides less accurate results than at higher power levels and therefore, may not enhance the response of the power range instrumentation channels. This same result, i.e., uncertain calibration, may be obtained without performing the surveillance. Since the results of the Surveillance are typically small adjustments, the change which allows nonperformance during low power does not significantly increase the consequences of an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure appropriate availability for the instrument channels considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The power range instrument channels provide no identifiable margin of safety since the difficulties of calibration to the calorimetric heat balance at low power are well known. In the conditions to be excluded from the Surveillance, the power range instrumentation is available, but calibration is recognized as generally less reliable than calibration at higher power levels. Therefore, the change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.3A L13

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

A Note is included which allows deferring the comparison and adjustment of the power range instrument channels of the RPS against the incore detectors while at low power levels. This Note recognizes that such a comparison at these low power levels provides inaccurate results. This change in applicability for this Surveillance does not result in any hardware changes. The power range monitors are not considered as initiators for any previously analyzed accidents. As such, the change does not significantly increase the probability of occurrence of any analyzed event. Performance of this Surveillance at low power levels does not provide accurate results and therefore, may not enhance the response of the power range instrumentation channels. This same result, i.e., uncertain calibration, may be obtained without performing the surveillance. Since the results of the Surveillance are typically small adjustments, the change which allows nonperformance during low power does not significantly increase the consequences of an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure appropriate availability for the instrument channels considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The power range instrument channels provide no identifiable margin of safety at low power since their calibration to the incore instrumentation does not provide accurate results. In the conditions to be excluded from the Surveillance, the power range instrumentation is available, but calibration is recognized as uncertain. Therefore, the change does not involve a significant reduction in the margin of safety.

3.3A L14

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

This change does not result in any hardware changes or changes in operating methods. The change removes an unnecessary additional performance of a surveillance which has been performed within its normal Frequency. Not performing the surveillance prior to startup would not affect any equipment which is assumed to be an initiator of any analyzed event. Further, since the surveillance continues to be performed on its normal Frequency, there is no impact on the capability of the system to perform its required safety function. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure adequate surveillance is performed to identify any degradation of the intermediate range instrumentation channel. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The margin of safety for intermediate range instrument channel is based on availability and capability of the instrument to perform its safety function. Since the normal periodic Frequency has been determined (through experience with CTS) to be adequate to confirm the availability and capability, the removal of an additional confirmatory check of the instrumentation does not impact that availability and capability. Therefore, this change does not involve a significant reduction in a margin of safety.

3.3A L15

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change does not result in any hardware changes or changes in operating methods. The change replaces a function test which is required to be performed "prior to startup" with a CHANNEL CALIBRATION on an 18 month Frequency. This change does not result in any hardware changes, and the source range instrumentation is not considered as the initiator of any previously analyzed accident. Therefore, the proposed change does not involve a significant increase in the probability of any accident previously evaluated. Additionally, neither the test, nor the test Frequency impact the operation of equipment or its response to any event. Therefore, the proposed change does not involve a significant increase in the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure adequate surveillance is performed to identify any degradation of the source range instrumentation channel. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The margin of safety for source range instrument channel is based on availability and capability of the instrument to perform its safety function. If the unit operates with only one "startup" per cycle, the Frequency for these surveillances is the same, but the calibration would be an additional requirement because it includes testing activities in addition to the functional test. Industry performance history of this type of instrumentation has demonstrated reliability of the equipment over an operating cycle. Therefore, a periodic Frequency of 18 months has been determined to be adequate to confirm the availability and capability. Therefore, this change does not involve a significant reduction in a margin of safety.

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3.3A L16

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Allowing changes in reactor coolant system (RCS) temperature during conditions when no source range or intermediate range neutron flux monitor may be operable while maintaining the required shutdown margin (SDM) will not alter the assumed initiation, and hence, will not significantly increase the probability of any evaluated event. The ITS contains actions that maintain the initial conditions assumed in the analyses. Because these Required Actions maintain the initial conditions assumed in the safety analyses, prevent the occurrence of evaluated events, and preserve the mitigatory response mechanisms should an event occur, the consequences of a postulated event from this condition would not be significantly increased.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure adequate verifications are performed to identify any potential change in reactivity. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will allow positive reactivity changes as a result of RCS temperature changes when no source range or intermediate range neutron flux monitor is operable. However, the ITS Required Actions limit such temperature-induced positive reactivity changes to within the limits of the minimum required SDM. Therefore, this change does not involve a significant reduction in a margin of safety.

ITS DISCUSSION OF DIFFERENCES

ITS Section 3.3A: Instrumentation - RPS

Note: The ITS Section 3.3A package addresses the following NUREG-1430 RSTS:

RSTS 3.3.1	RPS Instrumentation
RSTS 3.3.2	RPS Manual Reactor Trip
RSTS 3.3.3	RPS - Reactor Trip Module
RSTS 3.3.4	CRD Trip Devices
RSTS 3.3.9	Source Range Neutron Flux
RSTS 3.3.10	Intermediate Range Neutron Flux

1 NUREG 3.3.1 - The CTS flexibility to allow one Reactor Protection System (RPS) channel, containing inoperable functions, to be maintained in an untripped and unbypassed state is maintained. This flexibility is consistent with CTS 3.5.1.3. As modified, ITS 3.3.1 ACTION A will allow one channel containing inoperable function(s) to be maintained untripped and unbypassed. ITS 3.3.1 ACTION B will, in the event that two channels contain inoperable functions, require one of these channels to be tripped. Two options are available for dealing with the second inoperable channel. Either it will be bypassed or bypass of the two remaining OPERABLE channels will be prevented. This change is consistent with current license basis.

2 NUREG 3.3.1 - Condition C has been revised to specify that this Condition applies when three or more RPS channels are inoperable. This change was made to maintain requirements consistent with CTS Table 3.5.1-1 Column 5 and Note 1 which provide specific requirements for the inoperability of more than 2 channels.

Without this addition, entry into the ACTION requirements of ITS LCO 3.0.3 would be required if three channels of RPS contained inoperable functions. Entry into the Required Actions of ITS 3.3.1 Condition C rather than the ACTION requirements of LCO 3.0.3 is more appropriate because specific Required Actions which result in the unit exiting the unique Applicability for each RPS Function are provided in ITS 3.3.1. These Required Actions consistently result in the unit exiting the specific Applicability within a specific Completion Time. For example ITS LCO 3.0.3 ACTION requirements would not provide specific Completion Times for reducing THERMAL POWER to less than 45% RTP in the event three channel of the Main Turbine Trip function were inoperable.

Additionally, several of the RPS functions are required to be OPERABLE while in MODE 5. ITS LCO 3.0.3 is not applicable in MODE 5 and therefore would not require the unit to exit the MODES of Applicability for those Functions required OPERABLE in MODE 5. This change is consistent with TSTF-217, Rev 1.

3 NUREG 3.3.1 - Response time testing of the Reactor Protection System (RPS), i.e., NUREG SR 3.3.1.7, is not adopted in ITS. Testing of this type is not required by ANO-1 CTS. Deletion of these Surveillance Requirements maintains consistency with the current ANO-1 administrative control of these activities and neither removes any current requirement nor adds any additional requirement. This change is consistent with current license basis.

4 NUREG 3.3.1 - The Applicable MODES for Nuclear Overpower High Setpoint function and RCS High Pressure function have been expanded to include MODE 3 when not in shutdown bypass operation with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal. This change provides for requirements which are more restrictive than those provided by ANO-1 CTS. This additional Applicability is appropriate to ensure that the instrumentation required to initiate the insertion of any withdrawn CONTROL RODS is OPERABLE whenever CONTROL RODS are withdrawn or capable of withdrawal. The automatic insertion of any withdrawn CONTROL ROD is consistent with evaluations of abnormalities initiated from MODE 3 (although no specific analyses have been performed for MODE 3 events.)

This change in Applicability also requires the addition of Note (d) to NUREG-1430 Table 3.3.1-1 and appropriate changes to the Bases. This change is consistent with TSTF-218, as revised to reflect ANO-1 plant specific differences.

5 NUREG 3.3.1 - The Frequency of SR 3.3.1.2 has been changed from 24 hours to 96 hours and once within 24 hours after a THERMAL POWER change of ≥ 10% RTP in one direction. This Frequency replaces the CTS requirement to perform this calibration twice weekly under steady state conditions and daily under non-steady state operating conditions. The change from twice weekly to 96 hours has been made to provide for calibration at an interval essentially equivalent to CTS requirements while presenting the Frequency in a format consistent with that of the ITS. The requirement to perform this calibration daily under non-steady state conditions has been clarified with the site specific details of what constitutes non-steady state conditions. These changes provide requirements which maintain calibration on a Frequency similar to current requirements while presenting these requirements in a format consistent with NUREG-1430. Further, the adoption of 96 hour Frequency is acceptable based on operating experience which indicates that the deviation between the power range instruments and the calorimetric seldom exceeds 2% RTP even after a four day period with no adjustment to the power range instruments.

6 NUREG 3.3.1 - The specific details of performance of ITS SR 3.3.1.3 have been removed. These details provided methodology and acceptance criteria not contained in CTS. The removal of these details maintains requirements consistent with CTS. The details of this testing are currently contained in implementing procedures and will be retained there. This change neither adds any new requirements nor removes any existing requirement. This change is consistent with current license basis.

3.3.1-06

7 NUREG 3.3.1 - Reactor Protection System (RPS) CHANNEL FUNCTIONAL TEST requirements are contained in CTS Table 4.1-1. The Frequency of this testing is specified as monthly with the Bases providing implementation details for performing this testing on a rotational or staggered basis. ITS SR 3.3.1.4 is adopted with a 31 day Frequency. The details of performance of this testing on a rotational basis is presented in the Bases of the SR. This change is made to provide requirements consistent with CTS for this testing. No new requirements are added by this change and no existing requirements are removed. This change is consistent with current license basis.

8 NUREG 3.3.4 - ACTION C has been revised and presented in INSERT 3.3-11A. These changes were made to provide requirements which are appropriate for all Control Rod Drive (CRD) power supplies which contain Electronic Trip Assembly (ETA) Relays, including the auxiliary power supply, and to maintain flexibility consistent with CTS requirements.

Required Action C.1 was revised to allow transfer of a CONTROL ROD group to a power supply with an inoperable ETA relay which is open. This change was made to allow a CONTROL ROD group to be transferred to a power supply, and powered indefinitely from it, even if that power supply has one inoperable, open ETA relay. As presented in NUREG 3.3.4 Required Action C.1 would not have allowed the affected CONTROL ROD group to be transferred to a power supply with an inoperable, though open ETA. This change was made to retain the CTS allowance to operate indefinitely with a CONTROL ROD group being powered from a power supply with an inoperable, but open ETA relay.

ITS Required Action C.2 was added to allow a safety rod group which is being powered from the auxiliary power supply to be transferred back to its normal DC hold power supply in the event one of the ETA relays associated with the auxiliary power supply is inoperable. By design, the safety rod groups' normal power supplies do not contain ETA relays. The requirement in NUREG 3.3.4 Required Action C.1 to transfer the affected CONTROL ROD group to a power supply with OPERABLE ETA relays would not have been fulfilled by transferring an affected safety rod group back to its fully OPERABLE normal power supply. This change is being made to ensure that the inoperability of an ETA relay associated with the auxiliary power supply does not result in unnecessarily restrictive action requirements.

ITS Required Action C.3 was added to provide the flexibility to place the SCRs associated with an inoperable ETA relay in trip rather than the associated AC CRD trip breaker. Placing the SCRs associated with the inoperable ETA relay in trip accomplishes the design function of the ETA relay, which is to interrupt power, while allowing the associated AC CRD trip breaker to remain closed. This change is consistent with the requirements of CTS Table 3.5.1-1, Note 23.

The word "Trip" in NUREG 3.3.4 Required Action C.2 was changed to "Open" in ITS 3.3.4 Required Action C.4. This change is considered editorial in nature and is made to prevent possible confusion about the acceptable methods of accomplishing this requirement.

9 NUREG 3.3.9 - ITS LCO 3.3.9 presents the OPERABILITY requirements for the source range instruments. This LCO was modified to indicate that only one channel of instrumentation is required to be OPERABLE. This change was made to provide requirements consistent with those in CTS Table 3.5.1-1, RPS Functional Unit 4. Additionally, NUREG 3.3.9 ACTION A, which dealt with the inoperability of one of the two required instrument channels, was deleted to provide ACTIONS which are consistent with the requirements of the LCO.

3.3.9-01

The TS required source range instrument has a range from 0.1 to 10^5 count per second (cps). A wide range gamma metrics indication comes on scale at 10^{-8} % power, which equates to approximately 1 cps on the source range scale. Intermediate range power channels come on scale at 10^{-11} % amps, which equates to approximately 200 cps on the source range scale. The ANO-1 reactor generally achieves criticality at approximately 1000 cps on the source range scale. In summary, both the wide range gamma metrics indication and the intermediate range channels are available by the approximate power range that criticality is achieved. Therefore, the current license basis is acceptable since other diverse indications are available at the point of criticality. Regardless of the other indications that exist, the adoption of the NUREG 3.3.9 ACTION conservatively requires all control rods to be fully inserted when the required source range monitor is inoperable and the required intermediate range monitor is not sufficiently on scale. This change is consistent with current license basis.

10 NUREG 3.3.10 - ITS LCO 3.3.10 presents the OPERABILITY requirements for the intermediate range instruments. This LCO was modified to indicate that only one channel of instrumentation is required to be OPERABLE. This change was made to provide requirements consistent with those in CTS Table 3.5.1-1 RPS Functional Unit 3. Additionally, NUREG 3.3.10 ACTION A, which dealt with the inoperability of one of the two required instrument channels, was deleted to provide ACTIONS which are consistent with the requirements of the LCO.

3.3.10-01

The TS required source range instrument has a range from 0.1 to 10^5 count per second (cps). A wide range gamma metrics indication comes on scale at 10^{-8} % power, which equates to approximately 1 cps on the source range scale. Intermediate range monitors (IRM) come on scale at 10^{-11} % amps, which equates to approximately 200 cps on the source range scale. The ANO-1 reactor generally achieves criticality at approximately 1000 cps on the source range scale. In summary, both the wide range gamma metrics indication and the intermediate range channels are available by the approximate power range that criticality is achieved. Likewise, the wide range gamma metrics monitor remains on scale throughout the range of the IRMs. The Point of Adding Heat (POAH) is reached just above the top of the source range scale. Once the POAH is reached, reactor coolant temperature, steam bypass valve, and auxiliary feed pump responses also provide an indication of plant power. Therefore, the current license basis is acceptable since other diverse indications are available from the point of criticality through the power range. Regardless of the other indications that exist, the adoption of the NUREG 3.3.10 ACTION conservatively requires all control rods to be fully inserted when the required IRM is inoperable. This change is consistent with current license basis.

11 NUREG 3.3.9 - The Note modifying NUREG LCO 3.3.9 was not retained in the ITS. The current source range instrument channels at ANO-1 use a fission chamber detector. This design does not require that the high voltage be removed from these instruments to protect them from operation above 1.0 E -10 amp indicated on the intermediate range channels. This change is being made to provide requirements consistent with unit design features. This change is consistent with current license basis.

12. NUREG 3.3.9 - Conditions B and C were revised to remove the term "THERMAL POWER level" from each. The difference between Condition B and Condition C is whether the intermediate range neutron flux instrumentation channel indicates greater than or less than 1.0 E-10 amp. These instruments provide a relative indication of neutron flux and are not calibrated against heat balance power. The correlation between THERMAL POWER level and intermediate range neutron flux indication, especially when low in the intermediate range, is not easily determined. Intermediate range instrument channel indications in the range of 1.0 E-10 amp are generally indicative of reactor power levels below "the point of adding heat" and therefore not detectable as THERMAL POWER. By specifying that the requirements in Condition B and C are based on THERMAL POWER level rather than indicated neutron flux, the requirements of Conditions B and C were unnecessarily confusing.

13 NUREG 3.3.1 - The Allowable Value for Table 3.3.1-1, Function 5, is revised to retain the CTS as recently approved for Amendment 186. The Function 8 wording is similarly revised for consistency. This change is consistent with current license basis.

14 NUREG 3.3.10 - The CTS requirement to perform a CHANNEL FUNCTIONAL TEST on the intermediate range instrument channels has been retained in ITS. CTS Table 4.1-1, Item 5, requires this testing monthly ITS SR 3.3.10.2 has been adopted to provide a requirement to perform this testing with a Frequency of 31 days. This requirement to perform a CHANNEL FUNCTIONAL TEST on the intermediate range instrument channels is retained in ITS to maintain testing requirements consistent with the ANO-1 CTS.

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15 NUREG 3.3.9 - SR 3.3.9.3 and NUREG 3.3.10 – SR 3.3.10.3 have been relocated to the bases for SR 3.3.1.1, SR 3.3.9.1, and SR 3.3.10.1. The SRs provided verification that at least one decade of overlap exists between the source range and intermediate range instruments, and the intermediate range and power range instruments. These SRs are unnecessary in that they duplicate the requirements of the CHANNEL CHECKs of SR 3.3.1.1, SR 3.3.9.1, and SR 3.3.10.1. The CHANNEL CHECK envelops these requirements since a failure to achieve the expected overlap would constitute failure of the channel to meet the established "agreement criteria." By relocating overlap information to the bases, the agreement criteria can be established to provide appropriate flexibility in determining which channel(s) may be inoperable.

This change incorporates TSTF 264, Rev. 0 which affects the following specifications: NUREG Bases SR 3.3.1.1, NUREG 3.3.9 – SR 3.3.9.3, NUREG Bases SR 3.3.9.1, NUREG 3.3.10 – SR 3.3.10.3, and NUREG Bases SR 3.3.10.1.

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16 Not Used.

17 NUREG 3.3.10 - The Applicability has been changed to specify that the intermediate range instrument channel is required in MODE 2 and in MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal. The addition of "MODES 3, 4, and 5" to the second statement of the Applicability was made to maintain the CTS allowance provided by Table 3.5.1-1 Note 2. This Note, as applied at ANO-1, defined the upper limit of the applicable MODES for the required intermediate range instrument channel as being 10% indicated neutron power. Without the addition of the appropriate MODES, to the second statement of the Applicability for ITS 3.3.10, an intermediate range channel would be required at all times in MODE 1. This requirement is inconsistent with the design of the intermediate range instrument channels which is to provide indication of neutron power while operating at low power levels (MODE 2). The required indication of neutron power level is provided by the power range instruments while in MODE 1. This change is consistent with current license basis and generic traveler TSTF-291 .

18 NUREG 3.3.1 - The Allowable Value for the Reactor Coolant Pump to Power trip function in ITS Table 3.3.1-1 has been modified to indicate the ANO-1 unit specific value including the appropriate description of the reactor coolant pump combination. This change has been made to provide an accurate description of this Allowable Value which is consistent with the ANO-1 CTS and design function. This change is consistent with current license basis.

19 NUREG Bases 3.3.1 - Reactor Protection System (RPS) design at ANO-1 provides a separate bistable for the Nuclear Overpower Low Setpoint function and for the Nuclear Overpower High Setpoint function. The Low Setpoint bistable is inserted into the RPS channel trip string, along with the Shutdown Bypass RCS High Pressure bistable, when the channel is placed in shutdown bypass. The Bases of ITS 3.3.1 has been modified to describe this site specific design difference in the ANO-1 RPS. This change is consistent with current license basis.

20 NUREG 3.3.3 - Conditions B and C have been revised to specify that these Conditions also apply when two or more Reactor Protection System (RPS) Reactor Trip Modules (RTMs) are inoperable. This change was made to provide ACTION requirements which specifically remove the unit from the Applicability for this Specification.

Without this addition, entry into the ACTION requirements of ITS LCO 3.0.3 would be required if more than one RTM is inoperable. Entry into the Required Actions of ITS 3.3.3 Condition B or C (depending upon the current MODE), rather than the ACTION requirements of LCO 3.0.3, is more appropriate because specific Required Actions which result in the unit exiting the Applicability for LCO 3.3.3 are provided. These Required Actions result in the unit exiting the specific Applicability by either opening the Control Rod Drive (CRD) trip breakers or removing power from the CRD system within a specific Completion Time. ITS LCO 3.0.3 ACTION requirements would not require opening the CRD trip breakers or removing power from the CRD system, and therefore, would not result in exiting the Applicability of ITS 3.3.3.

Additionally, because the Applicability of ITS 3.3.3 includes operation in MODE 5 with the any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal, this change to Condition C will provide for appropriate ACTION requirements to exit the Applicability where ITS 3.0.3 would not. ITS LCO 3.0.3 is not applicable in MODE 5 and therefore would not require any ACTION while in MODE 5. This change is consistent with TSTF-217, Rev 1, as revised for consistency with the description of Condition B.

21 NUREG 3.3.1 Bases - The Bases have been generally revised to reflect unit specific design, analysis, and operating practices, or to provide descriptions consistent with other unit specific documents and terminology. These changes are consistent with current license basis or are editorial. Some specific revisions are as follows:

-Background-
The discussion of trip signals is revised for clarity. Two channels are required to initiate a reactor trip. However, two trip signals will not generate a reactor trip if they are in the same channel.

Discussion of CRD trip devices was revised to more clearly represent the actual configuration of the ANO-1 Control Rod Drive system power supplies. This clarification has been made to provide unit specific information to better describe the function and design of the electronic trip assembly (ETA) relays and other portions of the CRD system.

Discussion revised to omit unnecessary detail for "complete" testing of "all" RPS trip Functions since there may be minor portions of specific RPS functions which are not fully testable. For example, many surveillances exclude neutron detectors. Such exclusions would not constitute complete testing of all Functions.

Discussion of Main Turbine Automatic Stop Oil Pressure was revised to match ANO-1 unit specific design.

Discussion of an indicator light was added to the Channel Bypass discussion.

Potentially confusing discussion of methodology of determining Allowable Values was removed from Trip Setpoints/Allowable Value discussion.

-Applicable Safety Analyses, LCO, and Applicability-
Discussion revised to omit "at all time the reactor is critical" since this is not consistent with the Applicability of all RPS Functions. This change is consistent with TSTF-292.

Reference to the Main Steam Safety Valves (MSSVs) has been removed from the RCS High Pressure Function discussion. The NUREG Bases indicated that these valves functioned to prevent RCS overpressurization and thereby protect the RCS High Pressure Safety Limit (SL). Chapter 3 and 14 of the ANO-1 SAR do not credit the MSSVs as functioning to prevent exceeding RCS SL.

3.3.1-01

The specified Allowable Values are conservative with respect to Instrument Society of American Standard, ISA-S67-04, "Setpoints of Nuclear Safety Related Instrumentation Used in Nuclear Power Plants." Actual in-plant setpoints are equal to or conservative to the specified Allowable Values and include instrument uncertainties where appropriate. A discussion of the instrument uncertainty methodology employed may be found in letter dated May 10, 2000, approving Amendment 207 to the ANO-1 Operating License.

Functions 2, 3, and 6 are revised to identify that the Allowable Value does not consider harsh environmental conditions because the trip is assumed to occur prior to degraded conditions being reached, not because the associated functions are not required to mitigate accidents that create harsh conditions. This is consistent with the ANO-1 Safety Analyses.

Specific Allowable Values for the reactor coolant pump power monitors were not included in the Bases of Function 7 Reactor Coolant Pump to Power. This change was made to maintain these values under current administrative controls.

The ANO-1 unit specific terminology of "electromatic relief valve (ERV)" has been substituted for the generic term "power operated relief valve (PORV)." This change has been made to maintain the specific terminology which is consistent with other ANO-1 licensing basis documentation.

ANO-1 unit specific design information for the RPS trip on Main Turbine Trip function has been added. The system at ANO-1 provides for four main turbine oil pressure switches each providing input to a single RPS channel though a buffer device.

The statement regarding the utilization of the bypass trips to prevent unit conditions from reaching a point where actuation is necessary is not true for all the Functions listed.

A heading of "General Discussion" was added near the bottom of NUREG page B 3.3-20. This was done to separate the general information, which follows, from the preceding discussion which is specific to the Shutdown Bypass RCS Pressure Function. Additionally, the heading of NUREG page B 3.3-21 was appropriately changed due to this new heading.

Applicability discussion on NUREG page B 3.3-20 and B 3.3-21 was replaced with inserted discussion. This change was made to more clearly express the individual Applicability of the Functions in Table 3.3.1-1.

-Actions-
Discussion of ACTIONS D.1 through G.1 were revised to more accurately express the Conditions. This change has been made to clarify possibly confusing descriptions of these Conditions.

-Surveillance Requirements-
Bases SR 3.3.1.2 has been revised by the addition of unit specific details of the determination of calorimetric (heat balance) power.

22 NUREG 3.3.3 - The Note modifying NUREG SR 3.3.3.1 has been deleted. This Note allowed a delay of up to 8 hours for the entry into the Conditions and Required Actions for the performance of this surveillance. The performance of this SR, at ANO-1, does not render the reactor trip module (RTM) inoperable. During performance of the referenced CHANNEL FUNCTIONAL TEST, the coincidence logic network of the RTM is aligned for normal service and the RTM is capable of receiving trip commands from and issuing trip commands to the other three channels. Without the removal of the Note modifying SR 3.3.3.1, the possibility for confusion by the unit staff with regard to the OPERABILITY of the RTM during testing. Because no allowance, similar to this Note, exists in CTS, its removal from ITS maintains current requirements. Reference to the Note was additionally removed from the Bases of SR 3.3.3.1.

23 NUREG 3.3.9 Bases - The Bases have been generally revised to reflect unit specific design, analysis, and operating practices, or to provide descriptions consistent with other unit specific documents and terminology. Some specific revisions are as follows:

-Background- and -LCO-
Discussion has been revised to indicate ANO-1 unit specific design information. These changes have been made to provide accurate information appropriate to the ANO-1 design.

-Actions-
Discussion associated with ITS Required Actions A.1 through A.4 has been revised to more clearly describe the intent of these Required Actions. The NUREG Bases, as written, implied that these Required Actions would remove the unit from the Applicability of this LCO. These Required Actions do not remove the unit from the Applicability of the LCO, but rather provide actions to limit positive reactivity additions and to detect any changes in SDM. This change is consistent with TSTF-293.

Discussion was also added to provide guidance in the event that no indication of intermediate range flux is available, coincident with a loss of the required source range instrument channel. This discussion indicates that with no indication of intermediate range flux, ITS Required Actions A.1 through A.4 are applicable.

-Surveillance Requirements-
General discussion, in SR 3.3.9.1, was edited to match the specific design of the instrumentation to which this SR is applicable. Discussion of "transmitter...drift" is not appropriate due to the design of the source range instrument channels.

Discussion of performance of CHANNEL CHECKS, for off scale low current loop instrument channels, was removed from SR 3.3.9.1 discussion. This discussion was not appropriate based on the design of the source range instrument channels

24 NUREG 3.3.2 Bases - The Bases have been generally revised to reflect unit specific design, analysis, and operating practices, or to provide descriptions consistent with other unit specific documents and terminology. Some specific revisions are as follows:

-Applicability-
The word "only" was replaced with "primary" when describing the RPS safety function of tripping the CONTROL RODS. The ANO-1 unit specific design of the Reactor Protection System (RPS) is such that it provides inputs to other safety systems. Most notably, the Emergency Feedwater Initiation and Control (EFIC) system receives input from the RPS.

-Actions-
The specific Condition descriptions have been corrected for accuracy.

25 NUREG 3.3.3 Bases - The Bases have been generally revised to reflect unit specific design, analysis, and operating practices, or to provide descriptions consistent with other unit specific documents and terminology. Some specific revisions are as follows:

-Actions-
The specific Condition descriptions have been corrected for accuracy.

26 NUREG 3.3.4 Bases - The Bases have been generally revised to reflect unit specific design, analysis, and operating practices, or to provide descriptions consistent with other unit specific documents and terminology. Some specific revisions are as follows:

-Background -
Discussion of CRD trip devices was revised to more clearly represent the actual configuration of the ANO-1 Control Rod Drive system power supplies. This clarification has been made to provide unit specific information to better describe the function and design of the electronic trip assembly (ETA) relays and other portions of the CRD system.

-LCO-
Discussion of CRD trip devices was revised to more clearly represent the actual configuration of the ANO-1 Control Rod Drive system power supplies. This clarification has been made to provide unit specific information to better describe the function and design of the electronic trip assembly (ETA) relays and other portions of the CRD system.

-Actions-
The headings "Condition A" and "Condition B" were removed and the appropriate Required Action headings were moved to provide a format consistent with other sections.

The phrase "channel containing" was removed to ensure that this was not misinterpreted to mean that power was required to be removed from the RPS channel associated with the inoperable trip breaker.

Discussion previously under "Condition B" was revised to remove possibly confusing descriptions this Condition.

The specific Condition descriptions have been corrected for accuracy.

-Surveillance Requirements-
"AC" was replaced with "trip" in Bases SR 3.3.4.1 to ensure that this statement was not interpreted to mean that testing of the diverse trip features of the DC breakers was not required.

27 NUREG 3.3.10 Bases - The Bases have been generally revised to reflect unit specific design, analysis, and operating practices, or to provide descriptions consistent with other unit specific documents and terminology. Some specific revisions are as follows:

-Background-
Discussion was revised to indicate ANO-1 unit specific design information. These changes have been made to provide accurate information appropriate to the ANO-1 design.

-Surveillance Requirements-
General discussion, in SR 3.3.10.1, was edited to match the specific design of the instrumentation to which this SR is applicable. Discussion of "transmitter...drift" is not appropriate due to the design of the intermediate range instrument channels.

Discussion of performance of CHANNEL CHECKS, for off scale low current loop instrument channels, was added to SR 3.3.10.1 discussion. This discussion was appropriate based on the design of the intermediate range instrument channels

28 NUREG 3.3.9 - Incorporates TSTF-009, Rev. 1.

29 NUREG Bases 3.3.1 - Incorporates TSTF-019, Rev. 1.

3.3.1-06

30 NUREG SR 3.3.1.2, SR 3.3.1.3, SR 3.3.1.5, SR 3.3.1.6, Bases SR 3.3.1.2, Bases SR 3.3.1.3, Bases SR 3.3.1.5, and Bases SR 3.3.1.6 – Incorporates TSTF 342, Rev. 1. The term "recalibrated" in the 1st paragraph of NUREG Bases SR 3.3.1.3 is changed to "adjusted" to be consistent with the changes incorporated by TSTF 342, Rev. 1.

31 NUREG 3.3.1, 3.3.4, and associated Bases - The requirement to remove all power to the CRD system could be interpreted to include all control power and logic cabinet power since they are a part of this system. It is more appropriate to remove power from all CRD trip breakers. This action places the unit in condition where the LCO no longer applies. This change incorporates TSTF-211, which has been revised for grammatical correctness in the discussion of Required Actions B.1, B.2.1, and B.2.2. The revision of this generic change is considered to be editorial in nature.

32 NUREG 3.3.1 -NUREG SR 3.3.1.2 and SR 3.3.1.3 are revised such that the Note delays applicability for these SRs until some period of time "after THERMAL POWER is ≥ 20% RTP" rather than "after THERMAL POWER is ≥ 15% RTP." This change is based on current unit specific application of validity of the calorimetric heat balance at low powers. This change is also consistent with the similar "standard" requirements for the other PWR vendors, i.e., NUREG-1431 and NUREG-1432.

33 NUREG 3.3.3 & 3.3.4 - Reactor Protection System (RPS) CHANNEL FUNCTIONAL TEST requirements are contained in CTS Table 4.1-1. The Frequency of this testing is specified as quarterly with the Bases providing implementation details for performing this testing on a rotational or staggered basis. ITS SR 3.3.3.1 and SR 3.3.4.1 are adopted with a 92 day Frequency. The details of performance of this testing on a rotational basis is presented in the Bases of each SR. This change is made to provide requirements consistent with CTS for this testing. No new requirements are added by this change and no existing requirements are removed. This change is consistent with current license basis.

ANO-333

34 NUREG 3.3.9, 3.3.10, and associated bases – Incorporates TSTF-286, Revison 2.

 CTS

3.3 INSTRUMENTATION

3.3.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1 Four channels of RPS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE. Table 3.5.1-1

N/A

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One channel inoperable.	A.1 Place channel in bypass or trip. <INSERT 3.3-1A>	1 hour	3.5.1.3 1 Table 3.5.1-1 Note 6
B. Two channels inoperable.	B.1 Place one channel in trip. AND B.2 .1 Place second channel in bypass. <INSERT 3.3-1B>	1 hour 1 hour	3.5.1.3 3.5.1.3 1
Three or more channels inoperable OR C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the Condition referenced in Table 3.3.1-1 for the Function.	Immediately	2 Table 3.5.1-1 Column 5
D. As required by Required Action C.1 and referenced in Table 3.3.1-1.	D.1 Be in MODE 3. AND D.2 Open all CONTROL ROD drive (CRD) trip breakers.	6 hours 6 hours	Table 3.5.1-1 Note 1 edit

(continued)

<INSERT 3.3-1A>

			CTS
<u>OR</u>			
A.2	Prevent bypass of remaining channels.	1 hour	3.5.1.3

<INSERT 3.3-1B>

<u>OR</u>			
B.2.2	Prevent bypass of remaining channels.	1 hour	3.5.1.3

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME	
E. As required by Required Action C.1 and referenced in Table 3.3.1-1.	E.1	Open all CRD trip breakers.	6 hours	N/A
F. As required by Required Action C.1 and referenced in Table 3.3.1-1.	F.1	Reduce THERMAL POWER < ~~[45]~~% RTP. 45	6 hours	3.5.1.9.3
G. As required by Required Action C.1 and referenced in Table 3.3.1-1.	G.1	Reduce THERMAL POWER < ~~[15]~~% RTP. 10	6 hours	3.5.1.9.3

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.1-1 to determine which SRs apply to each RPS Function. N/A

SURVEILLANCE	FREQUENCY	
SR 3.3.1.1 Perform CHANNEL CHECK.	12 hours	Table 4.1-1 "Check"

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.3.1.2	-------------------NOTE-------------------- Not required to be performed until ~~[24]~~ hours after THERMAL POWER is ≥ ~~[15%]~~ 20% RTP. -- ~~Verify~~ Compare results of calorimetric heat balance ~~is~~ Calculation to ≤ [2]% RTP ~~greater than~~ power range channel output. Adjust power range channel output if ~~calorimetric exceeds power range channel output by ≥ [2]% RTP.~~ the absolute difference is >	96 hours AND Once within 24 hours after a THERMAL POWER change of ≥ 10% RTP	N/A 5 32 Table 4.1-1 Item 3 30
SR 3.3.1.3	-------------------NOTE-------------------- Not required to be performed until ~~[24]~~ hours after THERMAL POWER is ≥ ~~[15%]~~ 20% RTP. -- Compare results of ~~out of~~ core measured AXIAL POWER IMBALANCE ~~(API₀)~~ to incore measured AXIAL POWER IMBALANCE ~~(API₁) as follows:~~ ~~$(RTP/TP)(API_0 - API_1)$ = imbalance error~~ ~~Perform CHANNEL CALIBRATION if the absolute value of the imbalance error is ≥ [2]% RTP.~~ 1. Adjust the power range channel imbalance output if the absolute value of the imbalance error is ≥ 2% RTP.	31 days	32 N/A EDIT Table 4.1-1 Item 4 "Check" 30 6
SR 3.3.1.4	Perform CHANNEL FUNCTIONAL TEST.	31 days ~~[45] days on a STAGGERED TEST BASIS~~	7 Table 4.1-1 "Test"
~~SR 3.3.1.5~~	~~-------------------NOTE-------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.~~	~~[92] days~~	N/A 30

(continued)

RAI 3.3.1-06

CTS

RAI 3.3.1-06

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.~~6~~ 5	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	~~[18]~~ 30 months
~~SR 3.3.1.7~~	~~-------------------NOTE-------------------~~ ~~Neutron detectors are excluded from RPS RESPONSE TIME testing.~~ ~~--~~ ~~Verify that RPS RESPONSE TIME is within limits.~~	~~[18] months on a STAGGERED TEST BASIS~~

N/A

Table 4.1-1 Calibrate

3

RAI 3.3.1-06

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Nuclear Overpower —				
a. High Setpoint	1,2[a] 3 5[d] N/A 4	D Table 3.5.1-1 Note 1	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.5 ~~SR 3.3.1.7~~ SR 3.3.1.4 3 30	≤ ~~104.9~~ 104.9% RTP Table 2.3-1

Table 3.5.1-1 RPS Functional Unit 2 + Table 2.3-1

Table 3.3.1-1 Item 4 Table 3.3.1-1 Item 3

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breaker~~s~~ in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation. 4

(1)

RA1 3.3.1-06

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
b. Low Setpoint	2(b),3(b) 4(b),5(b)	E	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5 ~~SR 3.3.1.7~~	≤ 5% RTP
Table 2.3-1	N/A	N/A N/A	3 30	Table 2.3-1

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip ~~breakers~~ breaker in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation. 4

(2)

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. RCS High Outlet Temperature	1,2	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ 618 °F

Table 3.5.1-1 RPS Functional Unit 5 + Table 2.3-1

N/A

Table 3.5.1-1 Note 1

Table 4.1-1 Item 7 "Check"

Table 4.1-1 Item 7 "Calibrate"

Table 4.1-1 Item 7 "Test"

Table 2.3-1

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip ~~breakers~~ in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) With any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

4

(3)

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. RCS High Pressure	1,2[(a)], 3[(d)] (4)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 ~~SR 3.3.1.7~~ (3)	≤ [2355] psig (2355)
Table 3.5.1-1 RPS Functional Unit 8a + Table 2.3-1	N/A	Table 3.5.1-1 Note 1	Table 4.1-1 Item 8 "calibrate" Table 4.1-1 Item 8 "Test"	Table 2.3-1

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation. (4)

(4)

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. RCS Low Pressure	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 ~~SR 3.3.1.7~~	≥ ~~[1800]~~ 1800 psig

Table 3.5.1-1 RPS Functional Unit 8b + Table 2.3-1

N/A

Table 3.5.1-1 / Note 1

Table 4.1-1 Item 9 "check"

3

Table 4.1-1 Item 9 "calibrate"
Table 4.1-1 Item 9 "Test"

Table 2.3-1

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

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5

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. RCS Variable Low Pressure	1,2(a)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≥ ~~([11.59] * T_{out} - [5037.8]) psig~~
Table 3.5.1-1 RPS Functional Unit 6 + Table 2.3-1	N/A	Table 3.5.1-1 Note 1	Table 4.1-1 Item 11 "Check" Table 4.1-1 Item 11 "Calibrate" Table 4.1-1 Item 11 "Test"	Table 2.3-1

13

As specified in the COLR

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip ~~breakers~~ breaker in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

4

6

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
Table 3.5.1-1 RPS Functional Unit 1D + Table 2.3-1	N/A	Table 3.5.1-1 Note 1	Table 4.1-1 Item 13 "check" Table 4.1-1 Item 13 "test" Table 4.1-1 Item 13 "calibrate"	Table 2.3-1
6. Reactor Building High Pressure	1,2,3(c)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	~~≤ [4] psig~~ ≤18.7 psig

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip ~~breakers~~ breaker in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

(4)

(7)

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
Table 3.5.1-1 RPS Functional Unit 9 & Table 2.3-1	N/A	Table 3.5.1-1 Note 1	Table 4.1-1 Item 12 "check" Table 4.1-1 Item 12 "test" Table 4.1-1 Item 12 "calibrate"	Table 2.3-1
7. Reactor Coolant Pump to Power	1,2[(a)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 ~~SR 3.3.1.7~~ (3)	≤55 ~~[5]~~% RTP with one ~~2/2~~ pumps ~~operating~~ in each loop (18)

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation. (4)

(8)

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE	1,2(a)	D	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.5 SR 3.3.1.6 ~~SR 3.3.1.7~~	~~Nuclear Overpower RCS Flow and AXIAL POWER IMBALANCE setpoint envelope in COLR~~

Table 3.5.1-1 RPS Functional Unit 7 + Table 2.3-1

N/A

Table 3.5.1-1 Note 1

Table 4.1-1 Item 4 "Calibrate"

Table 4.1-1 Item 4 "Check"

Table 4.1-1 Item 10 "Check"

Table 4.1-1 Item ? "Calibrate"

Table 2.3-1

SR 3.3.1.4

30

3

As specified

the

13

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

4

(9)

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
9. Main Turbine Trip (Control Oil Pressure) edit	≥ [45]% RTP 25	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≥ [45] psig 40.5 N/A

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

4

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
10. Loss of Main Feedwater Pumps (Control Oil Pressure)	≥ [15]% RTP 20	G	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≥ [55] psig 55.5

N/A

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

4

(1)

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
11. Shutdown Bypass RCS High Pressure	$2^{(b)}, 3^{(b)}$ $4^{(b)}, 5^{(b)}$ N/A	E N/A	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 N/A	≤ [1720] psig 1720

Table 2.3-1

Table 2.3-1

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

4

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CTS

3.3 INSTRUMENTATION

3.3.2 Reactor Protection System (RPS) Manual Reactor Trip

LCO 3.3.2 The RPS Manual Reactor Trip Function shall be OPERABLE.

Table 3.5.1-1 Item 1

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with any ~~CONTROL ROD~~ control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

edit N/A

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Manual Reactor Trip Function inoperable.	A.1 Restore Function to OPERABLE status.	1 hour	N/A
B. Required Action and associated Completion Time not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Open all CRD trip breakers.	6 hours 6 hours	Table 3.5.1-1 Note 1
C. Required Action and associated Completion Time not met in MODE 4 or 5.	C.1 Open all CRD trip breakers.	6 hours	N/A

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.2.1 Perform CHANNEL FUNCTIONAL TEST.	Once prior to each reactor startup if not performed within the previous 7 days

Table 4.1-1 Item 44 "Test"

CTS

3.3 INSTRUMENTATION

3.3.3 Reactor Protection System (RPS)—Reactor Trip Module (RTM)

LCO 3.3.3 Four RTMs shall be OPERABLE. N/A

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with any control rod ~~CONTROL ROD~~ drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal. edit N/A

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME		
A. One RTM inoperable.	A.1.1 Open ~~Trip~~ the associated CRD trip breaker.	1 hour	EDIT	N/A
	OR			
	A.1.2 Remove power from the associated CRD trip breaker.	1 hour		N/A
	AND			
	A.2 Physically remove the inoperable RTM.	1 hour		N/A
B. Two or more RTMs inoperable in MODE 1, 2, or 3. OR Required Action and associated Completion Time not met in MODE 1, 2, or 3.	B.1 Be in MODE 3.	6 hours	20	N/A
	AND			
	B.2.1 Open all CRD trip breakers.	6 hours		N/A
	OR			
	B.2.2 Remove ~~all~~ power from ~~to~~ all ~~the~~ CRD ~~System.~~ trip breakers.	6 hours	31	N/A

(continued)

CTS

Two or more RTMs inoperable in MODE 4 or 5.
OR

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time not met in MODE 4 or 5.	C.1	Open all CRD trip breakers.	6 hours
	OR		
	C.2	Remove ~~all~~ power ~~to~~ them all ~~the~~ CRD ~~system.~~ trip breakers.	6 hours

20 N/A

31 N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	~~-----NOTE----- When an RTM is placed in an inoperable status solely for performance of this Surveillance, entry into associated Conditions and Required Actions may be delayed for up to 8 hours, provided at least two RTM channels are OPERABLE.~~ Perform CHANNEL FUNCTIONAL TEST.	92 days ~~[45] days on a STAGGERED TEST BASIS~~

22

33 Table 4.1-1 Item 1 "Test"

CTS

3.3 INSTRUMENTATION

3.3.4 ~~CONTROL ROD~~ Drive (CRD) Trip Devices

edit

LCO 3.3.4 The following CRD trip devices shall be OPERABLE: Table 3.5.1-1

a. Two AC CRD trip breakers; item 14A

b. Two DC CRD trip breaker pairs; and item 14B

c. Eight electronic trip assembly (ETA) relays. item 13

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with ~~when~~ any CRD trip breaker ~~is~~ in the closed position and the CRD System ~~is~~ capable of rod withdrawal.

edit
N/A

ACTIONS

NOTE

Separate Condition entry is allowed for each CRD trip device.

N/A

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. One or more CRD trip breaker(s) ~~(~~or breaker pair~~)~~ undervoltage or shunt trip Functions inoperable.	A.1	Open ~~Trip~~ the CRD trip breaker.	48 hours	edit Table 3.5.1-1 Note 25
	OR			
	A.2	Remove power from the CRD trip breaker.	48 hours	
B. One or more CRD trip breaker(s) ~~(~~for breaker pair~~)~~ inoperable for reasons other than those in Condition A.	B.1	Open ~~Trip~~ the CRD trip breaker.	1 hour	edit Table 3.5.1-1 Note 24, a.1
	OR			
	B.2	Remove power from the CRD trip breaker.	1 hour	Table 3.5.1-1 Note 24, a.2 3.5.1.6

(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
C. One or more required ETA relays inoperable. <INSERT 3.3-11A>	C.1 Transfer affected CONTROL ROD group to power supply with OPERABLE or open ETA relays. ~~OR~~ (8) C.~~2~~ 4 Open ~~Trip~~ corresponding AC CRD trip breaker.	1 hour 1 hour	-EDIT N/A N/A
D. Required Action and associated Completion Time not met in MODE 1, 2, or 3.	D.1 Be in MODE 3. AND D.2.1 Open all CRD trip breakers. OR D.2.2 Remove ~~all~~ power ~~to~~ from ~~the~~ all CRD ~~system.~~ trip breakers.	6 hours 6 hours 6 hours	N/A (31)
E. Required Action and associated Completion Time not met in MODE 4 or 5.	E.1 Open all CRD trip breakers. OR E.2 Remove ~~all~~ power ~~to~~ from all ~~the~~ CRD ~~system.~~ trip breakers.	6 hours 6 hours	N/A (31)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY	
SR 3.3.4.1 Perform CHANNEL FUNCTIONAL TEST.	~~31~~ 92 days	(33) Table 4.1-1 ITEMS 2 "Test" 62 "Test"

<INSERT 3.3-11A>

		CTS
OR		
C.2 Transfer affected CONTROL ROD group to a DC hold power supply.	1 hour	N/A
OR		
C.3 Place the SCRs associated with the inoperable ETA relay(s) in trip.	1 hour	Table 3.5.1-1 Note 23

CTS

3.3 INSTRUMENTATION

3.3.9 Source Range Neutron Flux

9

Table 3.5.1-1 Functional Unit 4

LCO 3.3.9 One ~~Two~~ source range neutron flux channel~~s~~ shall be OPERABLE.

~~-----NOTE-----~~
~~High voltage to detector may be de-energized above 1E-10 amp on intermediate range channels.~~
~~----------~~ 11

Table 3.5.1-1 Note 2

APPLICABILITY: MODES 2, 3, 4, and 5. N/A

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One source range neutron flux channel inoperable with THERMAL POWER level ≤ 1E-10 amp on the intermediate range neutron flux channels.~~	~~A.1 Restore channel to OPERABLE status.~~	~~Prior to increasing THERMAL POWER~~ 9
A. ~~B.~~ Required ~~Two~~ source range neutron flux channel~~s~~ inoperable ~~with THERMAL POWER level ≤ 1E-10 amp on the intermediate range neutron flux channels.~~	A.1 ~~B.1~~ Suspend operations involving positive reactivity changes.	Immediately N/A 12
	AND A.2 ~~B.2~~ Initiate action to insert all CONTROL RODS.	Immediately 34
	AND A.3 ~~B.3~~ Open CONTROL ROD drive trip breakers.	1 hour edit
	AND	

------ NOTE ------
Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SDM.

AND-333

(continued)

CTS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A ~~B~~. (continued)	A ~~B~~.4 Verify SDM ~~is~~ ~~≥ 1% Δk/k~~ to be within the limit provided in the COLR. 28	1 hour AND Once per 12 hours thereafter
B ~~C~~. Required ~~One or more~~ source range neutron flux channel~~(s)~~ inoperable with ~~THERMAL POWER level~~ > 1E-10 amp on the intermediate range neutron flux channel~~s~~. 12	B ~~C~~.1 Initiate action to restore ~~affected~~ required channel~~(s)~~ to OPERABLE status.	1 hour

N/A

Table 3.5.1-1
Note 3
9

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.9.2	-------------------NOTE-------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	~~1~~18~~4~~ months

Table 4.1-1
Item 6 "Check"

N/A

N/A

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.9.3	Verify at least one decade overlap with intermediate range neutron flux channels.	Once each reactor startup prior to source range counts exceeding 10^5 cps if not performed within the previous 7 days

15

CTS

3.3 INSTRUMENTATION

3.3.10 Intermediate Range Neutron Flux

LCO 3.3.10 One ~~Two~~ intermediate range neutron flux channel~~s~~ shall be OPERABLE.

10 Table 3.5.1-1 RPS Functional Unit 3.

APPLICABILITY: MODE 2,
MODES 3, 4, and 5 ~~When~~ with any control rod ~~CONTROL ROD~~ drive (CRD) trip breaker ~~is~~ in the closed position and the CRD System ~~is~~ capable of rod withdrawal.

Table 3.5.1-1 Note 2 N/A

17

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One channel inoperable.~~	~~A.1 Reduce THERMAL POWER to < 1E-10 amp.~~	~~2 hours~~
A. ~~B.~~ Required ~~Two~~ channel~~s~~ inoperable.	A.1 ~~B.1~~ Suspend operations involving positive reactivity changes.	Immediately
	AND	
	A.2 ~~B.2~~ Open CRD trip breakers.	1 hour

10

Table 3.5.1-1

------NOTE------
Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SDM.

34

N/A

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.10.1 Perform CHANNEL CHECK.	12 hours

Table 4.1-1 Item 5 "Check"

(continued)

<INSERT 3.3-25A> 14

<INSERT 3.3-25A>

CTS

SR 3.3.10.2	Perform CHANNEL FUNCTIONAL TEST.	31 days	Table 4.1-1 Item 5 "Test"

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.10.~~2~~ 3	--------NOTE-------- Neutron detectors are excluded from CHANNEL CALIBRATION. -------- Perform CHANNEL CALIBRATION.	~~[18]~~ months
~~SR 3.3.10.3~~	~~Verify at least one decade overlap with power range neutron flux channels.~~	~~Once each reactor startup prior to intermediate range indication exceeding 1E-6 amp if not performed within the previous 7 days~~

N/A

N/A

15

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