TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ALLOWABLE VALUE	APPLICABLE MODES	ACTION
7.	Overtemperature ΔT	3	2	2	See Table Notation (A)	1, 2	7
8.	Overpower ΔT	3	2	2	See Table Notation (B)	1, 2	7
9.	Pressurizer Pressure-Low (Above P-7)	3	2	2	≥ 1941 psig	1, 2	7
10.	Pressurizer Pressure-High	3	2	2	≤ 2389 psig	1, 2	7
11.	Pressurizer Water Level-High (Above P-7)	3	2	2	≤ 92.5% of instrument span	1, 2	7
12.	Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	≥ 89.8% of indicated loop flow	1	7
13.	Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	≥ 89.8% of indicated loop flow	1	7
14.	Steam Generator Water Level-Low-Low (Loop Stop Valves Open)	3/loop	2/loop	2/loop	≥ 19.6% of narrow range instrument span-each steam generator	1, 2	7

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	Functional Unit	Channel Check	Channel Calibration	Channel Functional Test	Modes in Which Surveillance Required
12.	Loss of Flow - Single Loop	S	R	Q	1
13.	Loss of Flow - Two Loops	S	R	Q	1
14.	Steam/Generator Water Level-Low-Low	S	R(16)(17)	Q(16)(17)	1, 2
15.	DELETED				
16.	Undervoltage-Reactor Coolant Pumps	N.A.	R	Q	1
17.	Underfrequency-Reactor Coolant Pumps	N.A.	R	Q	1
18.	Turbine Trip				
	a. Auto Stop Oil Pressure	N.A.	N.A.	S/U(1)	1, 2
	b. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	1, 2
19.	Safety Injection Input from ESF	N.A.	N.A.	R	1, 2
20.	Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	N.A.
21.	Reactor Trip Breaker	N.A.	N.A.	M(5,11) and S/U(1)	1, 2, 3(14), 4(14), 5(14)

TABLE 4.3-1 (Continued)

NOTATION (Continued)

(16) - If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

(17) - The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Nominal Trip Setpoint, or a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoint and the methodology used to determine the Nominal Trip Setpoint, the predefined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in a document incorporated by reference into the Updated Final Safety Analysis Report.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ALLOWABLE VALUE	APPLICABLE MODES	ACTION
7. AUXILIARY FEEDWATER						
a. Steam Gen. Water Level-Low-Low (Loop Stop Valves Open)						
i. Start Turbine Driven Pump	3/stm. gen.	2/stm. gen. any stm. gen.	2/stm. gen.	≥ 19.6% of narrow range instrument span each steam generator	1, 2, 3	14
ii. Start Motor Driven Pumps	3/stm. gen. any 2 stm. gen.	2/stm. gen. any 2 stm. gen.	2/stm. gen.	≥ 19.6% of narrow range instrument span each steam generator	1, 2, 3	14
b. Undervoltage-RCP (Start Turbine Driven Pump)	(3)-1/bus	2	2	≥ 71.2% rated RCP bus voltage	1	14
c. S.I. (Start All Auxiliary Feedwater Pumps)	See 1 above (all S.I. initiating functions and requirements)					
d. (Deleted)						
e. Trip of Main Feedwater Pumps (Start Motor Driven Pumps)	1/pump	1	1	Not Applicable	1, 2, 3	18

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
7.	AUXILIARY FEEDWATER				
	a. Steam Generator Water Level-Low-Low	S	R(2)(3)	Q(2)(3)	1, 2, 3
	b. Undervoltage-RCP	S	R	Q	1, 2
	c. S.I.	See 1 above (all SI surveillance requirements)			
	d. (Deleted)				
	e. Trip of Main Feedwater Pumps	N.A.	N.A.	R	1, 2, 3
8.	ESF INTERLOCKS				
	a. P-4	N.A.	N.A.	R	1, 2, 3
	b. P-11	N.A.	R	Q	1, 2, 3
	c. P-12	N.A.	R	Q	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train or logic channel shall be tested at least every other 31 days.

(2) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

(3) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Nominal Trip Setpoint, or a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoint and the methodology used to determine the Nominal Trip Setpoint, the predefined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in a document incorporated by reference into the Updated Final Safety Analysis Report.

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

The Nominal Trip Setpoint is based on the calculated total loop uncertainty per the plant specific methodology documented in the Licensing Requirements Manual. The setpoint methodology, used to derive the Nominal Trip Setpoints, is based upon combining all of the uncertainties in the channels. Inherent in the determination of the Nominal Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels should be capable of operating within the allowances of these uncertainty magnitudes. Occasional drift in excess of the allowance may be determined to be acceptable based on the other device performance characteristics. Device drift in excess of the allowance that is more than occasional, may be indicative of more serious problems and would warrant further investigation.

For certain Functional Units specified in Table 4.3-1 and Table 4.3-2 additional requirements are applied by Table Notes 16 and 2, respectively. If the "as found" value is found to be non-conservative with respect to the Allowable Value for the Functional Unit specified in Table 3.3-1 or 3.3-3, the channel is declared inoperable. If the "as found" value is found to be outside the two sided predefined acceptance criteria band, even if the "as found" setting is conservative with respect to the Allowable Value, Table 4.3-1 Note 16 or Table 4.3-2 Note 2 requires that an assessment of the channel performance is performed prior to returning the channel to service. The evaluation of channel performance will verify that the channel will continue to behave in accordance with design basis assumptions, and ensures confidence in the channel performance prior to returning the channel to service. If the "as found" trip setpoint value is non-conservative with respect to the Allowable Value or is found to be outside of the two sided predefined acceptance criteria band on either side of the Nominal Trip Setpoint, the affected channel is evaluated under the corrective action program.

For the Functional Units specified in Table 4.3-1 and Table 4.3-2 where Table Notes 17 and 3 respectively are applicable, Note 17 and Note 3 require the instrument channel setpoint to be reset to a value within the "as left" setpoint tolerance band on either side of the Nominal Trip Setpoint or to a value that is more conservative than the Nominal Trip Setpoint. The conservative direction is established by the direction of the inequality sign applied to the associated Allowable Value. Setpoint restoration and post-test verification assure that the assumptions in the plant setpoint methodology are satisfied in order to protect the safety analysis limits. If the channel can not be reset to a value within its "as left" setpoint tolerance band on either side of the Nominal Trip Setpoint, or to a value that is more conservative than the Nominal Trip Setpoint if

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

required based on plant conditions, the channel is declared inoperable and the applicable ACTION is entered.

Table 4.3-1 Note 16 and Note 17, and Table 4.3-2 Note 2 and Note 3 are applicable to specific instrument functions since changes to Allowable Values associated with these instrument functions were already under review by the NRC at the time the revised NRC setpoint criteria were documented and made available to the industry in an NRC letter to the Nuclear Energy Institute. Changes to the remaining instrument functions may be pursued after guidance endorsed by both the NRC and NEI is issued.

The "as found" and "as left" setpoint data for these specific Functional Units obtained during CHANNEL FUNCTIONAL TESTS or CHANNEL CALIBRATIONS are programmatically trended to demonstrate that the rack drift assumptions used in the plant setpoint methodology are valid. If the trending evaluation determines that a channel is performing inconsistent with the uncertainty allowances applicable to the periodic surveillance test being performed, the channel is evaluated under the corrective action program. If the channel is not capable of performing its specified safety function, it is declared inoperable.

The Engineered Safety Features Actuation System and Reactor Trip System Nominal Trip Setpoints specified in the Licensing Requirements Manual (LRM) are the nominal values* at which the instrumentation is set for each functional unit. An instrument setting is considered to be acceptable when the measured "as left" Setpoint is within the administratively controlled (±) calibration tolerance identified in plant procedures (which specifies the difference between the Allowable Value and Nominal Trip Setpoint). Additionally, a trip setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions provided that the ± calibration tolerance band remains the same and the allowable value is also adjusted accordingly in the conservative direction to meet the assumptions of the setpoint methodology.

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* With the exception of the Reactor Trip System Functional Unit number 17.B for the Turbine Stop Valve Position trip. The trip setpoint specified in the LRM for Functional Unit number 17.B is not a nominal value. The trip setpoint for this Functional Unit is adjusted to be consistent with the trip setpoint value specified in the LRM in lieu of adjusting the setpoint within an established calibration tolerance band.

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

Technical specifications are required by 10 CFR 50.36 to contain Limiting Safety System Settings (LSSS) defined by the regulation as "...settings for automatic protective devices...so chosen that automatic protective action will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytic Limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytic Limit therefore ensures that the SL is not exceeded.

For Functional Units in Tables 4.3-1 and 4.3-2 for which Table Note 16 and Table Note 2 respectively apply, the Nominal Trip Setpoints specified in the Licensing Requirements Manual are the LSSS. For Functional Units to which Table 4.3-1 Note 16 and Table 4.3-2 Table Note 2 are not applicable, the LSSS required by 10 CFR 50.36 are the Allowable Values specified in TS Tables 3.3-1 and 3.3-3. This definition of the LSSS is consistent with the guidance issued to the industry through correspondence with NEI (Reference NRC-NEI Letter dated September 7, 2005). The definition of LSSS values continues to be discussed between the industry and the NRC, and further modifications to these TS Bases will be implemented as guidance is provided. These values have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The various reactor trip circuits automatically open the reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides Reactor Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System.

The Reactor Trip System initiates a turbine trip signal whenever reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

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

The difference between T' (Overtemperature ΔT) or T" (Overpower ΔT) and the loop specific, indicated, full power T_{avg} shall be less than or equal to the T_{avg} allowances for such differences in the uncertainty calculations for these functions. In addition, T' and T" shall be less than or equal to the full power T_{avg} modeled in the safety analyses as an initial condition assumption; i.e., the numerical value specified in the COLR. In the event that the difference between a T' or T" set to the numerical value specified in the COLR and a loop specific, indicated, full power T_{avg} is greater than the T_{avg} allowances for such differences in the uncertainty calculations, T' or T" shall be reduced until the difference allowances in the uncertainty calculations are satisfied; i.e., T' or T" are set to a loop specific, full power value less than the numerical value specified in the COLR. These reductions in the values of T' and T" are consistent with the recommendations of Westinghouse Technical Bulletin ESBU-TB-96-07-RO, "Temperature Related Functions," 11/5/96.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low set point provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above the P-10 setpoint and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below the P-10 setpoint).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

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

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above the design DNBR limit for control rod drop accidents. At high power a single or multiple rod drop accident could cause flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. For those transients on which reactor trip on power range negative rate trip is not postulated, it is shown that the minimum DNBR is greater than the design DNBR limit.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor start-up. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at the trip setpoint unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to the trip setpoint unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in the COLR. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 3.3-1.

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

Overpower ΔT

The Overpower ΔT reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature ΔT protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above P-7, an automatic reactor trip will occur if the flow in any two loops drop below the trip setpoint. Above P-8, an automatic reactor trip will occur if the flow in any single loop drops below the trip setpoint.

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

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The trip setpoints assure a reactor trip signal is generated before the low flow trip set point is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 0.9 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip set point is reached shall not exceed 0.3 seconds.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-9. Each of the turbine trips provides turbine protection and reduces the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. This trip is provided to protect the core in the event of a LOCA. The ESF instrumentation channels which initiate a safety injection signal are shown in Table 3.3-3.

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

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide reactor core protection against DNB resulting from the opening of two or more pump breakers above P-7. These trips are blocked below P-7. The open/close position trips assure a reactor trip signal is generated before the low flow trip set point is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection System.

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions:

P-6 Above the setpoint P-6 allows the manual block of the Source Range reactor trip and de-energizing of the high voltage to the detectors. Below the setpoint Source Range level trips are automatically reactivated and high voltage restored.

P-7 Above the setpoint P-7 automatically enables reactor trips on low flow or coolant pump breaker open in more than one primary coolant loop, reactor coolant pump bus undervoltage and underfrequency, pressurizer low pressure and pressurizer high level. Below the setpoint the above listed trips are automatically blocked.

P-8 Above the setpoint P-8 automatically enables reactor trip on low flow in one or more primary coolant loops. Below the setpoint P-8 automatically blocks the above listed trip.

P-9 Above the setpoint P-9 automatically enables a reactor trip on turbine trip. Below the setpoint P-9 automatically blocks a reactor trip on turbine trip.

P-10 Above the setpoint P-10 allows the manual block of the Intermediate Range reactor trip and the low setpoint Power Range reactor trip; and automatically blocks the Source Range reactor trip and de-energizes the Source Range high voltage power. Below the setpoint the Intermediate Range reactor trip and the low setpoint Power Range reactor trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

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

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

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

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

The surveillance requirements for the Manual Trip Function, Reactor Trip Breakers and Reactor Trip Bypass Breakers are provided to reduce the possibility of an Anticipated Transient Without Scram (ATWS) event by ensuring OPERABILITY of the diverse trip features (Reference: Generic Letter 85-09).

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

ESF response times which include sequential operation of the RWST and VCT valves are based on values assumed in the Non-LOCA safety analyses and are provided in Section 3 of the Licensing Requirements Manual. These analyses take credit for injection of borated water. Initial borated water is supplied by the BIT, however, injection of borated water from the RWST is assumed not to occur until the VCT charging pump suction valves are closed following opening of the RWST charging pump suction valves. When sequential operation of the RWST and VCT valves is not included in the response times, the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response times will assure that the assumptions used for the LOCA and Non-LOCA analyses with respect to operation of the VCT and RWST valves are valid.

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

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

The Engineered Safety Feature Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates turbine trip, closes main feedwater valves on Tavg below setpoint, prevents the opening of the main feedwater valves which were closed by a safety injection or high steam generator water level signal, allows safety injection block so that components can be reset or tripped. Reactor not tripped - prevents manual block of safety injection.

P-11 Above the setpoint P-11 automatically reinstates safety injection actuation on low pressurizer pressure, automatically blocks steamline isolation on high steam pressure rate, enables safety injection and steamline isolation on low steamline pressure (with Loop Stop Valves Open), and enables auto actuation of the pressurizer PORVs.

Below the setpoint P-11 allows the manual block of safety injection actuation on low pressurizer pressure, allows manual block of safety injection and steamline isolation on low steamline pressure (with Loop Stop Valves Open) and enabling steamline isolation on high steam pressure rate, automatically disables auto actuation of the pressurizer PORVs unless the Reactor Vessel Over Pressure Protection System is in service.

P-12 Above the setpoint P-12 automatically reinstates an arming signal to the steam dump system. Below the setpoint P-12 blocks steam dump and allows manual bypass of the steam dump block to cooldown condenser dump valves.

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

Table 3.3-1 Action 2 has been modified by two notes. Note (4) allows placing the inoperable channel in the bypass condition for up to 4 hours while performing: a) routine surveillance testing of other channels, and b) setpoint adjustments of other channels when required to reduce the setpoint in accordance with other technical specifications. The 4 hour time limit is justified in accordance with WCAP-10271-P-A, Supplement 2, Revision 1, June 1990. Note (5) only requires SR 4.2.4 to be performed if a Power Range High Neutron Flux channel input to QPTR becomes inoperable. Failure of a component in the Power Range High Neutron Flux channel which renders the High Neutron Flux trip function inoperable may not affect the capability to monitor QPTR. As such, determining QPTR using the movable incore detectors once per 12 hours may not be necessary.

The following discussion pertains to Table 3.3-3, Functional Units 6.b and 6.c and the associated ACTION 34. The degraded voltage protection instrumentation system will automatically initiate the separation of the offsite power sources from the emergency buses. This action results in an automatic diesel generator start signal being generated as a direct result of the supply breakers opening between the normal and emergency buses. The failure of the degraded voltage protection system results in a loss of one of the automatic start signals for the diesel generator. Therefore, the ACTION statement requires the affected diesel generator to be declared inoperable if the required actions cannot be met within the specified time period.

The instrumentation functions that receive input from neutron detectors are modified by a note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 15% RATED THERMAL POWER. The power range neutron detector CHANNEL CALIBRATION is performed every 18 months but is not required for entry into MODE 2 or 1 on unit startup because the unit must be in at least MODE 1 to perform the test. The neutron detector CHANNEL CALIBRATION for the source range and intermediate range detectors consists of obtaining detector characteristics and performing an engineering evaluation of those characteristics. The intermediate range neutron detector CHANNEL CALIBRATION is performed every 18 months but is not required for entry into MODE 2 on unit startup because the unit must be in at least MODE 2 to perform the test. The source range neutron detector CHANNEL CALIBRATION is performed

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

every 18 months but is not required for entry into MODE 2 or 3 on unit shutdown because the unit must be in at least MODE 3 to perform the test. The P-6 permissive neutron detector CHANNEL CALIBRATION is performed in conjunction with the intermediate range neutron detectors. The overtemperature ΔT, P-8, P-9 and P-10 permissive neutron detector CHANNEL CALIBRATIONS are performed in conjunction with the power range neutron detectors.

Source Range Neutron Flux

The limiting condition for operation (LCO) requirement for the source range neutron flux trip function ensures that protection is provided against an uncontrolled rod cluster control assembly (RCCA) bank rod withdrawal accident from a subcritical condition during startup with the reactor trip breakers (RTBs) closed. This trip function provides redundant protection to the Power Range Neutron Flux-Low Setpoint and Intermediate Range Neutron Flux trip functions (See UFSAR Section 14.1.1). In MODES 3, 4, and 5, with the RTBs closed, administrative controls also prevent the uncontrolled withdrawal of rods. The nuclear instrumentation system (NIS) source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. In Modes 3, 4, and 5, with the reactor trip breakers closed, the source range detectors provide an automatic trip function with a setpoint in the shutdown range and the intermediate range detectors provide an automatic trip function with a setpoint in the power range. Therefore, the functional capability at the specified trip setpoint is assumed to be available.

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

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

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

In MODE 2 when below the P-6 setpoint during a reactor startup, the source range neutron flux trip must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and Power Range Neutron Flux-Low Setpoint trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS source range detectors are de-energized and not functional.

In MODE 3, 4, or 5 with the reactor shut down and with the control rod drive (CRD) system capable of rod withdrawal, the source range neutron flux trip function must also be OPERABLE. If the CRD system is not capable of rod withdrawal, the source range detectors are not required to trip the reactor. However, their monitoring function must be OPERABLE to monitor core neutron levels and provide indication of reactivity changes that may occur as a result of events like a boron dilution.

Suitable detectors used in place of primary source range neutron flux monitors are recognized as alternate detectors. Alternate detectors may be used in place of primary source range neutron flux monitors as long as the required neutron flux indication, high flux at shutdown alarm, and source range high neutron flux trip functions are provided.

ACTION 4

Item (a) applies to one inoperable source range neutron flux trip channel when in MODE 2, below the P-6 setpoint, and performing a reactor startup. With the unit in this condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

Item (b) applies to one inoperable source range neutron flux trip channel when in MODE 3, 4, or 5, with the RTBs closed and the CRD system capable of rod withdrawal. With the unit in this condition, the NIS source range performs the monitoring and protection functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to OPERABLE status. If the channel cannot be returned to an OPERABLE status, 1 additional hour

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

is allowed to open the RTBs. Once the RTBs are open, rod withdrawal is not possible and the unit enters ACTION 5. The allowance of 48 hours to restore the channel to OPERABLE status, and the additional hour to open the RTBs, are justified in WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.

Item (c) applies to two inoperable source range neutron flux trip channels when in MODE 2, below the P-6 setpoint, and performing a reactor startup, or in MODE 3, 4, or 5 with the RTBs closed and the CRD system capable of rod withdrawal. With the unit in this condition, below P-6, the NIS source range performs the monitoring and protection functions. With both source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, rod withdrawal is not possible and the unit enters ACTION 5.

ACTION 5

This ACTION applies when the required number of OPERABLE source range neutron flux channels is not met in MODE 3, 4, or 5 with the RTBs open. With the unit in this condition, the NIS source range performs the monitoring function. With less than the required number of source range channels OPERABLE, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation. However, a note applicable to this ACTION allows plant cooldown as long as the shutdown margin is adequate to account for the positive reactivity addition resulting from the temperature change. This ensures the core is controlled and the shutdown margin requirements are satisfied for all applicable events. In addition to suspension of positive reactivity additions, the valve(s) that controls the addition of unborated water to the RCS must be closed within 1 hour. The isolation of unborated water sources will preclude a boron dilution accident.

Also, the shutdown margin (SDM) must be verified within 1 hour and once every 12 hours thereafter as per SR 4.1.1.1.1 or 4.1.1.2, SDM verification. With no source range channels OPERABLE, core protection is severely reduced. Verifying the SDM within 1 hour allows sufficient time to perform the calculations and determine that the SDM requirements are met. The SDM must also be verified once per 12 hours thereafter to ensure that the core reactivity has not changed. Item (a) precludes any positive reactivity additions; therefore, core reactivity should not be increasing, and a 12 hour frequency is adequate. This does not include xenon decay which is accounted for in the shutdown margin surveillance. The completion times of within 1 hour and once per 12 hours are based on operating experience in performing the ACTIONS and the knowledge that unit conditions will change slowly.

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

SOURCE RANGE NEUTRON FLUX

SURVEILLANCE REQUIREMENTS (SR)

CHANNEL CHECK

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

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar 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. A CHANNEL CHECK for a single channel involves a qualitative assessment of the channel indication to verify the channel is operating in the approximate range for the expected plant conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is key to 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 the channel instrument uncertainties, including indication and readability. If a channel is outside the match criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The frequency is based on operating experience that demonstrates channel failure is rare. Thus, performance of the CHANNEL CHECK ensures that undetected overt channel failure is limited to 12 hours. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

When the control rods are fully inserted and are not capable of withdrawal, inadvertent control rod withdrawal is not a concern and one source range detector can adequately monitor the core.

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

CHANNEL FUNCTIONAL TEST

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

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure the entire channel will perform the intended function. Setpoints must be within the Allowable Values. The frequency of 92 days is justified for certain channels in WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.

This surveillance is modified by a Note that specifies testing when below P-6 and is clarified to address the transition from MODE 2 to MODE 3. A transition into MODE 3 with the reactor trip breakers closed is often made for a short period of time during plant shutdown. During a normal shutdown, the reactor trip breakers are opened shortly after entering MODE 3. The transition time in MODE 3 from when the reactor trip breakers are closed to when they are opened is less than the time required to perform the CHANNEL FUNCTIONAL TEST prior to entering MODE 3. Therefore, an allowance to enter MODE 3 without first performing the source range CHANNEL FUNCTIONAL TEST is warranted.

When performing the CHANNEL FUNCTIONAL TEST for manual initiation functions, the injection of a simulated signal into the channel as close to the primary sensor as practicable is accomplished by manually operating the function's manual switch(es).

CHANNEL CALIBRATION

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

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

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

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ALLOWABLE VALUE	APPLICABLE MODES	ACTION
7.	Overtemperature ΔT	3	2	2	See Table Notation (A)	1, 2	7
8.	Overpower ΔT	3	2	2	See Table Notation (B)	1, 2	7
9.	Pressurizer Pressure-Low (Above P-7)	3	2	2	≥ 1941 psig**	1, 2	7
10.	Pressurizer Pressure-High	3	2	2	≤ 2379 psig	1, 2	7
11.	Pressurizer Water Level-High (Above P-7)	3	2	2	≤ 92.5% of instrument span	1, 2	7
12.	Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	≥ 89.6% of indicated loop flow	1	7
13.	Loss of Flow - Two Loop (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	≥ 89.6% of indicated loop flow	1	7
14.	Steam Generator Water Level-Low-Low	3/loop	2/loop	2/loop	≥ 20% of narrow range instrument span-each steam generator	1, 2	7

** Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are ≥ 2 seconds for lead and ≤ 1 second for lag. Channel calibration shall ensure that these time constants are adjusted to those values.

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	Channel Check	Channel Calibration	Channel Functional Test	Modes in Which Surveillance Required
12. Loss of Flow - Single Loop (Above P-8)	S	R	Q	1
13. Loss of Flow - Two Loop (Above P-7 and Below P-8)	S	R	Q	1
14. Steam/Generator Water Level-Low-Low	S	R(16)(17)	Q(16)(17)	1, 2
15. DELETED.				
16. Undervoltage-Reactor Coolant Pumps (Above P-7)	N.A.	R	Q	1
17. Underfrequency-Reactor Coolant Pumps (Above P-7)	N.A.	R	Q	1
18. Turbine Trip (Above P-9)				
A. Emergency Trip Header Low Pressure	N.A.	R	S/U(1)	1, 2
B. Turbine Stop Valve Closure	N.A.	R	S/U(1)	1, 2
19. Safety Injection Input from ESF	N.A.	N.A.	R	1, 2
20. Reactor Coolant Pump Breaker Position Trip (Above P-7)	N.A.	N.A.	R	N.A.
21. Reactor Trip Breaker	N.A.	N.A.	M(5, 11) and S/U(1)	1, 2, 3(14), 4(14), 5(14)

TABLE 4.3-1 (Continued)

NOTATION (Continued)

(16) - If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

(17) - The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Nominal Trip Setpoint, or a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoint and the methodology used to determine the Nominal Trip Setpoint, the predefined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in a document incorporated by reference into the Updated Final Safety Analysis Report.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ALLOWABLE VALUE	APPLICABLE MODES	ACTION
5. TURBINE TRIP & FEEDWATER ISOLATION						
a. Automatic Actuation Logic and Actuation Relays	2	1	2	N.A.	1, 2	42
b. Steam Generator Water Level--High-High, P-14	3/loop	2/loop in any operating loop	2/loop in each operating loop	≤ 92.7% of narrow range instrument span	1, 2, 3	14
c. Safety Injection	See Item 1 above for all Safety Injection initiating functions and requirements.					
6. LOSS OF POWER						
a. 4.16kv Emergency Bus						
1. Undervoltage (Trip Feed)	2/4.16kv Bus	2/4.16kv Bus	2/4.16kv Bus	≥ 71.2% of rated Bus Voltage with a 1 ± 0.1 second time delay	1, 2, 3, 4	33
2. Undervoltage (Start Diesel)	1/4.16kv Bus	1/4.16kv Bus	1/4kv Bus	≥ 71.2% of rated Bus Voltage, 20 cycles ± 2 cycles	1, 2, 3, 4	33
b. 4.16kv Emergency Bus (Degraded Voltage)	2/4.16kv Bus	2/Bus	2/Bus	≥ 93.1% of rated Bus Voltage with a 90 ± 5 second time delay	1, 2, 3, 4	34

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ALLOWABLE VALUE	APPLICABLE MODES	ACTION
6. LOSS OF POWER (Continued)						
c. 480 Volt Emergency Bus (Degraded Voltage)	2/480v Bus	2/Bus	2/Bus	≥ 93.1% of rated Bus Voltage with a 90 ± 5 second time delay	1, 2, 3, 4	34
7. AUXILIARY FEEDWATER[3]						
a. Automatic Actuation Logic and Actuation Relays	2	1	2	N.A.	1, 2, 3	42
b. Steam Gen. Water Level--Low-Low						
1. Start Turbine Driven Pump	3/stm. gen.	2/stm. gen. any stm. gen.	2/stm. gen.	≥ 20% of narrow range instrument span	1, 2, 3	14
2. Start Motor Driven Pumps	3/stm. gen.	2/stm. gen. any 2 stm. gen.	2/stm. gen.	≥ 20% of narrow range instrument span	1, 2, 3	14
c. Undervoltage-RCP (Start Turbine Driven Pump)	(3)-1/bus	2	2	≥ 71.2% of rated bus voltage	1, 2	14

(3) Manual initiation is included in Specification 3.7.1.2.

TABLE 4.3-2 (Continued)

ENGINEERING SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual Initiation				
1. Individual	N.A.	N.A.	R	1, 2, 3
2. System	N.A.	N.A.	R	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	$M^{(1)}$	1, 2, 3
c. Containment Pressure--Intermediate-High-High	S	R	Q	1, 2, 3
d. Steamline Pressure--Low	S	R	Q	1, 2, 3
e. Steamline Pressure Rate-High Negative	S	R	Q	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	$M^{(1)}$	1, 2, 3
b. Steam Generator Water Level--High-High, P-14	S	$R^{(2)(3)}$	$Q^{(2)(3)}$	1, 2, 3
c. Safety Injection	See Functional Unit 1 above for all Safety Injection Surveillance Requirements.			

TABLE 4.3-2 (Continued)

ENGINEERING SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
6. LOSS OF POWER				
a. 4.16kv Emergency Bus				
1. Undervoltage (Trip Feed)	N.A.	R	Q	1, 2, 3, 4
2. Undervoltage (Start Diesel)	N.A.	R	Q	1, 2, 3, 4
b. 4.16kv Emergency Bus (Degraded Voltage)	N.A.	R	Q	1, 2, 3, 4
c. 480v Emergency Bus (Degraded Voltage)	N.A.	R	Q	1, 2, 3, 4
7. AUXILIARY FEEDWATER(4)				
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	M(1)	1, 2, 3
b. Steam Generator Water Level-Low-Low				
1. Start Turbine Driven Pump	S	R(2)(3)	Q(2)(3)	1, 2, 3
2. Start Motor Driven Pumps	S	R(2)(3)	Q(2)(3)	1, 2, 3

(4) Manual initiation is included in Specification 3.7.1.2.

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train or logic channel shall be tested at least every other 31 days.

(2) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

(3) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Nominal Trip Setpoint, or a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoint and the methodology used to determine the Nominal Trip Setpoint, the predefined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in a document incorporated by reference into the Updated Final Safety Analysis Report.

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

The Nominal Trip Setpoint is based on the calculated total loop uncertainty per the plant specific methodology documented in the Licensing Requirements Manual. The setpoint methodology, used to derive the Nominal Trip Setpoints, is based upon combining all of the uncertainties in the channels. Inherent in the determination of the Nominal Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels should be capable of operating within the allowances of these uncertainty magnitudes. Occasional drift in excess of the allowance may be determined to be acceptable based on the other device performance characteristics. Device drift in excess of the allowance that is more than occasional, may be indicative of more serious problems and would warrant further investigation.

For certain Functional Units specified in Table 4.3-1 and Table 4.3-2 additional requirements are applied by Table Notes 16 and 2, respectively. If the "as found" value is found to be non-conservative with respect to the Allowable Value for the Functional Unit specified in Table 3.3-1 or 3.3-3, the channel is declared inoperable. If the "as found" value is found to be outside the two sided predefined acceptance criteria band, even if the "as found" setting is conservative with respect to the Allowable Value, Table 4.3-1 Note 16 or Table 4.3-2 Note 2 requires that an assessment of the channel performance is performed prior to returning the channel to service. The evaluation of channel performance will verify that the channel will continue to behave in accordance with design basis assumptions, and ensures confidence in the channel performance prior to returning the channel to service. If the "as found" trip setpoint value is non-conservative with respect to the Allowable Value or is found to be outside of the two sided predefined acceptance criteria band on either side of the Nominal Trip Setpoint, the affected channel is evaluated under the corrective action program.

For the Functional Units specified in Table 4.3-1 and Table 4.3-2 where Table Notes 17 and 3 respectively are applicable, Note 17 and Note 3 require the instrument channel setpoint to be reset to a value within the "as left" setpoint tolerance band on either side of the Nominal Trip Setpoint or to a value that is more conservative than the Nominal Trip Setpoint. The conservative direction is established by the direction of the inequality sign applied to the associated Allowable Value. Setpoint restoration and post-test verification assure that the assumptions in the plant setpoint methodology are satisfied in order to protect the safety analysis limits. If the channel can not be reset to a value within its "as left" setpoint tolerance band on either side of the Nominal Trip Setpoint, or to a value that is more conservative than the Nominal Trip Setpoint if

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

required based on plant conditions, the channel is declared inoperable and the applicable ACTION is entered.

Table 4.3-1 Note 16 and Note 17, and Table 4.3-2 Note 2 and Note 3 are applicable to specific instrument functions since changes to Allowable Values associated with these instrument functions were already under review by the NRC at the time the revised NRC setpoint criteria were documented and made available to the industry in an NRC letter to the Nuclear Energy Institute. Changes to the remaining instrument functions may be pursued after guidance endorsed by both the NRC and NEI is issued.

The "as found" and "as left" setpoint data for these specific Functional Units obtained during CHANNEL FUNCTIONAL TESTS or CHANNEL CALIBRATIONS are programmatically trended to demonstrate that the rack drift assumptions used in the plant setpoint methodology are valid. If the trending evaluation determines that a channel is performing inconsistent with the uncertainty allowances applicable to the periodic surveillance test being performed, the channel is evaluated under the corrective action program. If the channel is not capable of performing its specified safety function, it is declared inoperable.

The Engineered Safety Features Actuation System and Reactor Trip System Nominal Trip Setpoints specified in the Licensing Requirements Manual (LRM) are the nominal values* at which the instrumentation is set for each functional unit. An instrument setting is considered to be acceptable when the measured "as left" setpoint is within the administratively controlled (±) calibration tolerance identified in plant procedures (which specifies the difference between the Allowable Value and Nominal Trip Setpoint). Additionally, a trip setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions provided that the ± calibration tolerance band remains the same and the Allowable Value is also adjusted accordingly in the conservative direction to meet the assumptions of the setpoint methodology.

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* With the exception of the Reactor Trip System Functional Unit number 17.b for the Turbine Stop Valve Position trip. The trip setpoint specified in the LRM for Functional Unit 17.b is not a nominal value. The trip setpoint for this Functional Unit is adjusted to be consistent with the trip setpoint value specified in the LRM in lieu of adjusting the setpoint within an established calibration tolerance band.

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

Technical specifications are required by 10 CFR 50.36 to contain Limiting Safety System Settings (LSSS) defined by the regulation as "...settings for automatic protective devices...so chosen that automatic protective action will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytic Limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytic Limit therefore ensures that the SL is not exceeded.

For Functional Units in Tables 4.3-1 and 4.3-2 for which Table Note 16 and Table Note 2 respectively apply, the Nominal Trip Setpoints specified in the Licensing Requirements Manual are the LSSS. For Functional Units to which Table 4.3-1 Note 16 and Table 4.3-2 Table Note 2 are not applicable, the LSSS required by 10 CFR 50.36 are the Allowable Values specified in TS Tables 3.3-1 and 3.3-3. This definition of the LSSS is consistent with the guidance issued to the industry through correspondence with NEI (Reference NRC-NEI Letter dated September 7, 2005). The definition of LSSS values continues to be discussed between the industry and the NRC, and further modifications to these TS Bases will be implemented as guidance is provided. These values have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The various reactor trip circuits automatically open the reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides Reactor Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System.

The Reactor Trip System initiates a turbine trip signal whenever reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

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

The difference between T' (Overtemperature ΔT) or T" (Overpower ΔT) and the loop specific, indicated, full power T_{avg} shall be less than or equal to the T_{avg} allowances for such differences in the uncertainty calculations for these functions. In addition, T' and T" shall be less than or equal to the full power T_{avg} modeled in the safety analyses as an initial condition assumption; i.e., the numerical value specified in the COLR. In the event that the difference between a T' or T" set to the numerical value specified in the COLR and a loop specific, indicated, full power T_{avg} is greater than the T_{avg} allowances for such differences in the uncertainty calculations, T' or T" shall be reduced until the difference allowances in the uncertainty calculations are satisfied; i.e., T' or T" are set to a loop specific, full power value less than the numerical value specified in the COLR. These reductions in the values of T' and T" are consistent with the recommendations of Westinghouse Technical Bulletin ESBU-TB-96-07-RO, "Temperature Related Functions," 11/5/96.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low setpoint provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above the P-10 setpoint) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below the P-10 setpoint).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

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

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above the design DNBR limit for control rod drop accidents. At high power a multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than the design DNBR limit.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at the trip setpoint unless manually blocked when P-6 becomes active. The intermediate range channels will initiate a reactor trip at a current level proportional to the trip setpoint unless manually blocked when P-10 becomes active. Although no explicit credit was taken for operation of the Source Range Channels in the accident analyses, operability requirements in the Technical Specifications will ensure that the Source Range Channels are available to mitigate the consequences of an inadvertent control bank withdrawal in MODES 3, 4 and 5.

Overtemperature ΔT

The overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit, thermowell, and RTD response time delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for transport, thermowell, and RTD response time delays from the core to RTD output indication. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in the COLR. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notation in Table 3.3-1.

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

Overpower ΔT

The Overpower ΔT reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature ΔT protection, and provides a backup to the High Neutron Flux Trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for transport, thermowell, and RTD response time delays from the core to RTD output indication. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam line breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Release."

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Pressure trip is automatically blocked by P-7; and on increasing power, automatically reinstated by P-7.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. On decreasing power, the pressurizer high water level trip is automatically blocked by P-7; and on increasing power, automatically reinstated by P-7. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

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

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above P-7, an automatic reactor trip will occur if the flow in any two loops drop below the trip setpoint. Above P-8, an automatic reactor trip will occur if the flow in any single loop drops below the trip setpoint.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The trip setpoints assure a reactor trip signal is generated before the low flow trip setpoint is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip setpoint is reached shall not exceed 0.6 seconds.

On decreasing power, the Undervoltage and Underfrequency Reactor Coolant Pump bus trips are automatically blocked by P-7; and on increasing power, reinstated automatically by P-7.

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

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-9. Each of the turbine trips provides turbine protection and reduces the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. This trip is provided to protect the core in the event of a LOCA. The ESF instrumentation channels which initiate a safety injection signal are shown in Table 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide reactor core protection against DNB resulting from the opening of two or more pump breakers above P-7. These trips are blocked below P-7. The open/close position trips assure a reactor trip signal is generated before the low flow trip setpoint is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection System.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 Above the setpoint P-6 allows the manual block of the Source Range reactor trip and de-energizing of the high voltage to the detectors. Below the setpoint source range level trips are automatically reactivated and high voltage restored.

P-7 Above the setpoint P-7 automatically enables reactor trips on low flow or coolant pump breaker open in more than one primary coolant loop, reactor coolant pump bus undervoltage and underfrequency, pressurizer low pressure and pressurizer high level. Below the setpoint the above listed trips are automatically blocked.

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

P-8 Above the setpoint P-8 automatically enables reactor trip on low flow in one or more primary coolant loops. Below the setpoint P-8 automatically blocks the above listed trip.

P-9 Above the setpoint P-9 automatically enables a reactor trip on turbine trip. Below the setpoint P-9 automatically blocks a reactor trip on turbine trip.

P-10 Above the setpoint P-10 allows the manual block of the Intermediate Range reactor trip and the low setpoint Power Range reactor trip; and automatically blocks the Source Range reactor trip and de-energizes the Source Range high voltage power. Below the setpoint the Intermediate Range reactor trip is automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.