ATTACHMENT 1

Dresden Station Unit 2

Proposed Technical Specification Changes

Revised Pages: 42
43
55
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New Page: 61a

8010240 210

INSTRUMENTATION THAT INITIATES ROD BLOCK

Table 3.2.3

Minimum No. of Operable Inst. Channels Per Trip System (1)	Instrument	Trip Level Setting
1	APRM upscale (flow bias) (7)	$\leq(.65W_D + 43)\ \frac{LTPF}{TPF}$ (2)
1	APRM upscale (refuel and Startup/Hot Standby mode)	$\leq$12/125 full scale
2	APRM downscale (7)	$\geq$3/125 full scale
1	Rod block monitor upscale (flow bias) (7)	$\leq$(.65W + 42) (2)
1	Rod block monitor downscale (7)	$\geq$5/125 full scale
3	IRM downscale (3)	$\geq$5/125 full scale
3	IRM upscale	$\leq$108/125 full scale
3	IRM detector not fully inserted in the core	
2(5)	SRM detector not in startup position	(4)
2(5) (6)	SRM upscale	$\leq 10^5$ counts/sec.
1	Scram Discharge Volume Water Level - High	25 gal.
1	Scram Trip Bypassed	N/A

TABLE 4.2.1

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS INSTRUMENTATION, ROD BLOCKS, AND ISOLATIONS

Instrument Channel	Instrument Functional Test (2)	Calibration (2)	Instrument Check (2)
ECCS INSTRUMENTATION			
1. Reactor Low-Low Water Level	(1)	Once/3 Months	Once/Day
2. Drywell High Pressure	(1)	Once/3 Months	None
3. Reactor Low Pressure	(1)	Once/3 Months	None
4. Containment Spray Interlock			
a. 2/3 Core Height	(1)	Once/3 Months	None
b. Containment High Pressure	(1)	Once/3 Months	None
5. Low Pressure Core Cooling Pump Discharge	(1)	Once/3 Months	None
6. Undervoltage Emergency Bus	Refueling Outage	Refueling Outage	None
7. Sustained High Reactor Pressure	(1)	Once/3 Months	None
ROD BLOCKS			
1. APRM Downscale	(1) (3)	Once/3 Months	None
2. APRM Flow Variable	(1) (3)	Refueling Outage	None
3. APRM Upscale (Startup/Hot Standby)	(2) (3)	(2) (3)	(2)
4. IRM Upscale	(2) (3)	(2) (3)	(2)
5. IRM Downscale	(2) (3)	(2) (3)	(2)
6. IRM detector not fully inserted in the core	(2)	N/A	None
7. RBM Upscale	(1) (3)	Refueling Outage	None
8. RBM Downscale	(1) (3)	Once/3 Months	None
9. SRM Upscale	(2) (3)	(2) (3)	(2)
10. SRM Detector Not in Startup Position	(2) (3)	(2) (3)	(2)
11. Scram Instrument Volume Level - High	Once/3 months	N/A	None
12. Scram Trip Bypassed	Refueling Outage	N/A	None
MAIN STEAM LINE ISOLATION			
1. Steam Tunnel High Temperature	Refueling Outage	Refueling Outage	None
2. Steam Line High Flow	(1)	Once/3 Months	Once/Day
3. Steam Line Low Pressure	(1)	Once/3 Months	None
4. Steam Line High Radiation	(1) (3)	Once/3 Months (4)	Once/Day

3.3 LIMITING CONDITION FOR OPERATION	4.3 SURVEILLANCE REQUIREMENT
b. The control rod directional control valves for inoperable control rods shall be disarmed electrically and the control rods shall be in such positions that Specification 3.3.A.1 is met. c. Control rod drives which are fully inserted and electrically disarmed shall not be considered inoperable. d. Control rods with scram times greater than those permitted by Specification 3.3.C are inoperable, but if they can be moved with control rod drive pressure, they need not be disarmed electrically if Specification 3.3.A.1 is met for each position of these rods. e. During reactor power operation, the number of inoperable control rods shall not exceed eight.	3. The scram discharge volume vent and drain valves shall be verified open at least once per 31 days. These valves may be closed intermittently for testing under administrative control. At least once each Refueling Outage, the scram discharge volume vent and drain valves will be demonstrated to: a. Close within 15 seconds after receipt of a signal for control rods to scram, and b. Open when the scram signal is reset.

indicative of a generic control rod drive problem and the reactor will be shutdown. Also, if damage within the control rod drive mechanism and, in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWR's. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

3. The operability of the scram discharge volume vent and drain valves assures the proper venting and draining of the volume. This ensures that water accumulation does not occur which would cause an early termination of control rod movement during a full core scram. These specifications provide for the periodic verification that the valves are open and for testing of these valves under reactor scram conditions during each Refueling Outage.

B. Control Rod Withdrawal

1. Control rod dropout accidents as discussed in the SAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement would provide cause for suspecting a rod to be uncoupled and stuck. Restricting recoupling verifications to power levels above 20% provides assurance that a rod drop during a recoupling verification would not result in a rod drop accident more severe than analyzed.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 6.6.1 of the SAR, and the design evaluation is given in Section 6.6.3. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. Control rod withdrawal and insertion sequences are established to assure that the maximum in-sequence individual control rod or control rod sequences which are withdrawn could not be worth enough to result in a peak fuel enthalpy of 280 cal/gm if they were to drop out of the core in the manner defined for the Rod Drop Accident.[3] These sequences are developed prior to operation

of the unit following any refueling outage and the requirement that an operator follow these sequences is backed up by the operation of the RWM. These sequences are developed to limit reactivity worths of control rods and, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content, 425 cal/gm, at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in Reference 1.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2 and 14.2,1.4 of the Safety Analysis Report. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident.

ATTACHMENT 2

Dresden Station Unit 3

Proposed Technical Specification Changes:

Revised Pages:	42
	43
	55
	62
New Page:	61a

INSTRUMENTATION THAT INITIATES ROD BLOCK

Table 3.2.3

Minimum No. of Operable Inst. Channels Per Trip System (1)	Instrument	Trip Level Setting
1	APRM upscale (flow bias) (7)	$\leq (.65W_D + 43) \left(\frac{FRP}{MFPLD}\right)$ (2)
1	APRM upscale (refuel and Startup/Hot Standby mode)	$\leq$12/125 full scale
2	APRM downscale (7)	$\geq$3/125 full scale
1	Rod block monitor upscale (flow bias) (7)	$\leq$(.65W + 42) (2)
1	Rod block monitor downscale (7)	$\geq$5/125 full scale
3	IRM downscale (3)	$\geq$5/125 full scale
3	IRM upscale	$\leq$108/125 full scale
3	IRM detector not fully inserted in the core	
2(5)	SRM detector not in startup position	(4)
2(5) (6)	SRM upscale	$\leq 10^5$ counts/sec.
1	Scram Discharge Volume Water Level - High	25 gal.
1	Scram Trip Bypassed	N/A

TABLE 4.2.1

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS INSTRUMENTATION, ROD BLOCKS, AND ISOLATIONS

Instrument Channel	Instrument Functional Test (2)	Calibration (2)	Instrument Check (2)
ECCS INSTRUMENTATION			
1. Reactor Low-Low Water Level	(1)	Once/3 Months	Once/Day
2. Drywell High Pressure	(1)	Once/3 Months	None
3. Reactor Low Pressure	(1)	Once/3 Months	None
4. Containment Spray Interlock			
a. 2/3 Core Height	(1)	Once/3 Months	None
b. Containment High Pressure	(1)	Once/3 Months	None
5. Low Pressure Core Cooling Pump Discharge	(1)	Once/3 Months	None
6. Undervoltage Emergency Bus	Refueling Outage	Refueling Outage	None
7. Sustained High Reactor Pressure	(1)	Once/3 Months	None
ROD BLOCKS			
1. APRM Downscale	(1) (3)	Once/3 Months	None
2. APRM Flow Variable	(1) (3)	Refueling Outage	None
3. APRM Upscale (Startup/Hot Standby)	(2) (3)	(2) (3)	(2)
4. IRM Upscale	(2) (3)	(2) (3)	(2)
5. IRM Downscale	(2) (3)	(2) (3)	(2)
6. IRM detector not fully inserted in the core	(2)	N/A	None
7. RBM Upscale	(1) (3)	Refueling Outage	None
8. RBM Downscale	(1) (3)	Once/3 Months	None
9. SRM Upscale	(2) (3)	(2) (3)	(2)
10. SRM Detector Not in Startup Position	(2) (3)	(2) (3)	(2)
11. Scram Instrument Volume Level - High	Once/3 months	N/A	None
12. Scram Trip Bypassed	Refueling Outage	N/A	None
MAIN STEAM LINE ISOLATION			
1. Steam Tunnel High Temperature	Refueling Outage	Refueling Outage	None
2. Steam Line High Flow	(1)	Once/3 Months	Once/Day
3. Steam Line Low Pressure	(1)	Once/3 Months	None
4. Steam Line High Radiation	(1) (3)	Once/3 Months (4)	Once/Day

3.3 LIMITING CONDITION FOR OPERATION	4.3 SURVEILLANCE REQUIREMENT
b. The control rod directional control valves for inoperable control rods shall be disarmed electrically and the control rods shall be in such positions that Specification 3.3.A.1 is met. c. Control rod drives which are fully inserted and electrically disarmed shall not be considered inoperable. d. Control rods with scram times greater than those permitted by Specification 3.3.C are inoperable, but if they can be moved with control rod drive pressure, they need not be disarmed electrically if Specification 3.3.A.1 is met for each position of these rods. e. During reactor power operation, the number of inoperable control rods shall not exceed eight.	3. The scram discharge volume vent and drain valves shall be verified open at least once per 31 days. These valves may be closed intermittently for testing under administrative control. At least once each Refueling Outage, the scram discharge volume vent and drain valves will be demonstrated to: a. Close within 15 seconds after receipt of a signal for control rods to scram, and b. Open when the scram signal is reset.

indicative of a generic control rod drive problem and the reactor will be shutdown. Also, if damage within the control rod drive mechanism and, in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWR's. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

3. The operability of the scram discharge volume vent and drain valves assures the proper venting and draining of the volume. This ensures that water accumulation does not occur which would cause an early termination of control rod movement during a full core scram. These specifications provide for the periodic verification that the valves are open and for testing of these valves under reactor scram conditions during each Refueling Outage.

B. Control Rod Withdrawal

1. Control rod dropout accidents as discussed in Reference 6 can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement would provide cause for suspecting a rod to be uncoupled and stuck. Restricting recoupling verifications to power levels above 20% provides assurance that a rod drop during a recoupling verification would not result in a rod drop accident.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 6.6.1 of the SAR, and the design evaluation is given in Section 6.6.3. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod sequences which are withdrawn could not be worth enough to cause the rod drop accident design limit of 280 cal/gm to be exceeded if they were to drop out of the core in the manner defined for the Rod Drop Accident. (3) These sequences are developed prior to initial operation.

of the unit following any refueling outage and the requirement that an operator follow these sequences is backed up by the operation of the RWM or a second qualified station employe. These sequences are developed to limit reactivity worths of control rods and, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content, 425 cal/gm, at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in Reference 1.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2 and 14.2,1.4 of the Safety Analysis Report. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident.

ATTACHMENT 3

Quad Cities Station Unit 1

Proposed Technical Specification Changes

Revised Pages: 3.2/4.2-14
3.2/4.2-16
3.3/4.3-3
3.3/4.3-9

TABLE 3.2-3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum Number of Operable or Tripped Instrument Channels per Trip System(1)	Instrument	Trip Level Setting
2	APRM upscale (flow bias)(7)	≤0.650W + 43(2)
2	APRM upscale (Refuel and Startup/Hot Standby mode)	≤12/125 full scale
2	APRM downscale(7)	≥3/125 full scale
1	Rod block monitor upscale (flow bias)(7)	≤0.650W + 42(2)
1	Rod block monitor downscale(7)	≥3/125 full scale
3	IRM downscale (3) (8)	≥3/125 full scale
3	IRM upscale(8)	≤108/125 full scale
2(5)	SRM detector not in Startup position(4)	≥2 feet below core center-line
3	IRM detector not in Startup position(8)	≥2 feet below core center-line
2(5), (6)	SRM upscale	≤10^5 counts/sec
2(5)	SRM downscale(9)	≥10^2 counts/sec
1	High water level in scram discharge volume (SDV)	≤25 gallons
1	SDV high water level scram trip bypassed	NA

Notes

1. For the Startup/Hot Standby and Run positions of the reactor mode selector switch, there shall be two operable or tripped trip systems for each function except the SRM rod blocks. IRM upscale and IRM downscale need not be operable in the Run position, APRM downscale, APRM upscale (flow biased), RBM upscale, and RBM downscale need not be operable in the Startup/Hot Standby mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to 7 days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than 7 days the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.
2. W is the reactor recirculation loop flow in percent. Trip level setting is in percent of rated power (2511 MWt).
3. IRM downscale may be bypassed when it is on its lowest range.
4. This function is bypassed when the count rate is ≥100 CPS.
5. One of the four SRM inputs may be bypassed.
6. This SRM function may be bypassed in the higher IRM ranges (ranges 8, 9, and 10) when the IRM upscale rod block is operable.
7. Not required to be operable while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.
8. This IRM function occurs when the reactor mode switch is in the Refuel or Startup/Hot Standby position.
9. This trip is bypassed when the SRM is fully inserted.

TABLE 4.2-1

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS INSTRUMENTATION, ROD BLOCKS, AND ISOLATIONS(7)

Instrument Channel	Instrument Functional Test(2)	Calibration(2)	Instrument Check(2)
ECCS Instrumentation			
1. Reactor low-low water level	(1)	Once/3 months	Once/day
2. Drywell high pressure	(1)	Once/3 months	None
3. Reactor low pressure	(1)	Once/3 months	None
4. Containment spray interlock			
a. 2/3 core height	(1)	Once/3 months	None
b. Containment pressure	(1)	Once/3 months	None
5. Low-pressure core cooling pump discharge	(1)	Once/3 months	None
6. Undervoltage 4-kV essential	Refueling outage	Refueling outage	None
Rod Blocks			
1. APRM downscale	(1) (3)	Once/3 months	None
2. APRM flow variable	(1) (3)	Refueling outage	None
3. IRM upscale	(5) (3)	(5) (3)	None
4. IRM downscale	(5) (3)	(5) (3)	None
5. RBM upscale	(1) (3)	Refueling outage	None
6. RBM downscale	(1) (3)	Once/3 months	None
7. SRM upscale	(5) (3)	(5) (3)	None
8. SRM detector not in startup position	(5) (3)	(6)	None
9. IRM detector not in startup position	(5)	(6)	None
10. SRM downscale	(5) (3)	(5) (3)	None
11. High water level in scram discharge volume (SDV)	Once/3 months	Not applicable	None
12. SDV high level trip bypassed	Refueling outage	Not applicable	None
Main Steamline Isolation			
1. Steam tunnel high temperature	Refueling outage	Refueling outage	None
2. Steamline high flow	(1)	Once/3 months	Once/day
3. Steamline low pressure	(1)	Once/3 months	None
4. Steamline high radiation	(1) (4)	Refueling outage	Once/day
5. Reactor low low water level	(1)	Once/3 months	Once/day
RCIC Isolation			
1. Steamline high flow	Once/3 months	Once/3 months	None
2. Turbine area high temperature	Refueling outage	Refueling outage	None
3. Low reactor pressure	Once/3 months	Once/3 months	None

3. The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

   a. Control rod withdrawal sequences shall be established so that maximum reactivity that could be added by dropout of any increment of any one control blade would not make the core more than 0.013 Δk supercritical.

   b. Whenever the reactor is in the Startup/Hot Standby or Run mode below 20% rated thermal power, the rod worth minimizer shall be operable. A second operator or qualified technical person may be used as a substitute for an inoperable rod worth minimizer which fails after withdrawal of at least 12 control rods to the fully withdrawn position. The rod worth minimizer may also be bypassed for low power physics testing to demonstrate the shutdown margin requirements of Specification 3.3.A if a nuclear engineer is present and verifies the step-by-step rod movements of the test procedure.

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second and these SRM's are fully inserted.

5. During operation with limiting control rod patterns, as determined by the nuclear engineer, either:

   a. both RBM channels shall be operable.

   b. control rod withdrawal shall be blocked; or

3. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified after loading the sequence.

   Prior to the start of control rod withdrawal towards criticality, the capability of the rod worth minimizer to properly fulfill its function shall be verified by the following checks:

   a. The RWM computer online diagnostic test shall be successfully performed.

   b. Proper annunciation of the selection error of one out-of-sequence control rod shall be verified.

   c. The rod block function of the RWM shall be verified by withdrawing the first rod as an out-of-sequence control rod no more than to the block point.

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

6. The scram discharge volume vent and drain valves shall be verified open at least once per 31 days. These valves may be closed intermittently for testing under administrative control. At least once each Refueling Outage, the scram discharge volume vent and drain valves will be demonstrated to:

   a. Close within 15 seconds after receipt of a signal for control rods to scram, and

   b. Open when the scram signal is reset.

b. an end-of-cycle delayed neutron fraction of 0.005,

c. a beginning-of-life Doppler reactivity feedback,

d. the rod scram insertion rate shown in Specification 3.3.C,

e. the maximum possible rod drop velocity of 3.11 fps,

f. the design accident and scram reactivity shape function, and

g. the moderator temperature at which criticality occurs.

In most cases the worth of insequence rods or rod segments will be substantially less than 0.013 Δk. Further, the addition of 0.013 Δk worth of reactivity, as a result of a rod drop and in conjunction with the actual values of the other important accident analysis parameters described above, would most likely result in a peak fuel enthalpy substantially less than 280 cal/g design limit. However, the 0.013 Δk limit is applied in order to allow room for future reload changes and ease of verification without repetitive technical specification changes.

Should a control drop accident result in a peak fuel energy content of 280 cal/g, fewer than 660 (7 x 7) fuel rods are conservatively estimated to perforate. This would result in an offsite dose well below the guideline value of 10 CFR 100. For 8 x 8 fuel, fewer than 850 rods are conservatively estimated to perforate, with nearly the same consequences as for the 7 x 7 fuel case because of the rod power differences.

The rod worth minimizer provides automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences (reference SAR Section 7.9). It serves as a backup to procedural control of control rod worth. In the event that the rod worth minimizer is out of service when required, a licensed operator or other qualified technical employee can manually fulfill the control rod pattern conformance function of the rod worth minimizer. In this case, the normal procedural controls are backed up by independent procedural controls to assure conformance.

4. The source range monitor (SRM) system performs no automatic safety system function, i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. This is needed for knowledgeable and efficient reactor startup at low neutron levels. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's is provided as an added conservatism.

5. The rod block monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator, who withdraws control rods according to a written sequence. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists. During reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one of more fuel rods with MCPR's less than 1.07. During use of such patterns, it is judged that testing of the RBM system to assure its operability prior to withdrawal of such rods will assure that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns.

6. The operability of the Scram Discharge Volume vent and drain valves assures the proper venting and draining of the Volume, so that water accumulation in the Volume does not occur. These specifications provide for the periodic verification that the valves are open, and for the testing of these valves under reactor scram conditions during each Refueling Outage.

ATTACHMENT 4

Quad Cities Station Unit 2

Proposed Technical Specification Changes

Revised Pages: 3.2/4.2-14
3.2/4.2-16
3.3/4.3-3
3.3/4.3-9

TABLE 3.2-3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum Number of Operable or Tripped Instrument Channels per Trip System(1)	Instrument	Trip Level Setting
2	APRM upscale (flow bias)(7)	$\leq [0.650W_D + 43^{(2)}] \frac{FRP}{MFLPD}$
2	APRM upscale (Refuel and Startup/Hot Standby mode)	≤12/125 full scale
2	APRM downscale(7)	≥3/125 full scale
1	Rod block monitor upscale (flow bias)(7)	$\leq 0.650W_D + 42^{(2)}$
1	Rod block monitor downscale(7)	≥3/125 full scale
3	IRM downscale (3) (8)	≥3/125 full scale
3	IRM upscale(8)	≤108/125 full scale
2(5)	SRM detector not in Startup position(4)	≥2 feet below core center-line
3	IRM detector not in Startup position(8)	≥2 feet below core center-line
2(5) (6)	SRM upscale	$\leq 10^5$ counts/sec
2(5)	SRM downscale(9)	$\geq 10^2$ counts/sec
1	High water level in scram discharge volume (SDV)	≤25 gallons
1	SDV high water level scram trip bypassed	NA

Notes

1. For the Startup/Hot Standby and Run positions of the reactor mode selector switch, there shall be two operable or tripped trip systems for each function except the SRM rod blocks. IRM upscale and IRM downscale need not be operable in the Run position, APRM downscale, APRM upscale (flow biased), and RBM downscale need not be operable in the Startup/Hot Standby mode. The RBM upscale need not be operable at less than 30% rated thermal power. One channel may be bypassed above 30% rated thermal power provided that a limiting control rod pattern does not exist. For systems with more than one channel per trip system, if the first column cannot be met for one of the two trip systems, this condition may exist for up to 7 days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than 7 days the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

2. W_D is the percent of drive flow required to produce a rated core flow of 98 million lb/hr. Trip level setting is in percent of rated power (2511 MWt).

3. IRM downscale may be bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is ≥100 CPS.

5. One of the four SRM inputs may be bypassed.

6. This SRM function may be bypassed in the higher IRM ranges (ranges 8, 9, and 10) when the IRM upscale rod block is operable.

7. Not required to be operable while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.

8. This IRM function occurs when the reactor mode switch is in the Refuel or Startup/Hot Standby position.

9. This trip is bypassed when the SRM is fully inserted.

TABLE 4.2-1

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS INSTRUMENTATION, ROD BLOCKS, AND ISOLATIONS(7)

Instrument Channel	Instrument Functional Test(2)	Calibration(2)	Instrument Check(2)
ECCS Instrumentation			
1. Reactor low-low water level	(1)	Once/3 months	Once/day
2. Drywell high pressure	(1)	Once/3 months	None
3. Reactor low pressure	(1)	Once/3 months	None
4. Containment spray interlock			
a. 2/3 core height	(1)	Once/3 months	None
b. Containment pressure	(1)	Once/3 months	None
5. Low-pressure core cooling pump discharge	(1)	Once/3 months	None
6. Undervoltage 4-kV essential	Refueling outage	Refueling outage	None
Rod Blocks			
1. APRM downscale	(1) (3)	Once/3 months	None
2. APRM flow variable	(1) (3)	Refueling outage	None
3. IRM upscale	(5) (3)	(5) (3)	None
4. IRM downscale	(5) (3)	(5) (3)	None
5. RBM upscale	(1) (3)	Refueling outage	None
6. RBM downscale	(1) (3)	Once/3 months	None
7. SRM upscale	(5) (3)	(5) (3)	None
8. SRM detector not in startup position	(5) (3)	(6)	None
9. IRM detector not in startup position	(5)	(6)	None
10. SRM downscale	(5) (3)	(5) (3)	None
11. High water level in scram discharge volume (SDV)	Once/3 months	Not applicable	None
12. SDV high level trip bypassed	Refueling outage	Not applicable	None
Main Steamline Isolation			
1. Steam tunnel high temperature	Refueling outage	Refueling outage	None
2. Steamline high flow	(1)	Once/3 months	Once/day
3. Steamline low pressure	(1)	Once/3 months	None
4. Steamline high radiation	(1) (4)	Refueling outage	Once/day
5. Reactor low low water level	(1)	Once/3 months	Once/day
RCIC Isolation			
1. Steamline high flow	Once/3 months	Once/3 months	None
2. Turbine area high temperature	Refueling outage	Refueling outage	None
3. Low reactor pressure	Once/3 months	Once/3 months	None

3. The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

   a. Control rod withdrawal sequences shall be established so that maximum reactivity that could be added by dropout of any increment of any one control blade would be such that the rod drop accident design limit of 280 cal/gm. is not exceeded.

   b. Whenever the reactor is in the Startup/Hot Standby or Run mode below 20% rated thermal power, the rod worth minimizer shall be operable. A second operator or qualified technical person may be used as a substitute for an inoperable rod worth minimizer which fails after withdrawal of at least 12 control rods to the fully withdrawn position. The rod worth minimizer may also be bypassed for low power physics testing to demonstrate the shutdown margin requirements of Specification 3.3.A if a nuclear engineer is present and verifies the step-by-step rod movements of the test procedure.

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second and these SRM's are fully inserted.

5. During operation with limiting control rod patterns, as determined by the nuclear engineer, either:

   a. both RBM channels shall be operable,

   b. control rod withdrawal shall be blocked; or

3. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified after loading the sequence.

   Prior to the start of control rod withdrawal towards criticality, the capability of the rod worth minimizer to properly fulfill its function shall be verified by the following checks:

   a. The RWM computer online diagnostic test shall be successfully performed.

   b. Proper annunciation of the selection error of one out-of-sequence control rod shall be verified.

   c. The rod block function of the RWM shall be verified by withdrawing the first rod as an out-of-sequence control rod no more than to the block point.

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

6. The scram discharge volume vent and drain valves shall be verified open at least once per 31 days. These valves may be closed intermittently for testing under administrative control. At least once each Refueling Outage, the scram discharge volume vent and drain valves will be demonstrated to:

   a. Close within 15 seconds after receipt of a signal for control rods to scram, and

   b. Open when the scram signal is reset.

b. the delayed neutron fraction chosen for the bounding reactivity curve

c. a beginning-of-life Doppler reactivity feedback

d. scram times slower than the Technical Specification rod scram insertion rate (Section 3.3.c.1)

e. the maximum possible rod drop velocity of 3.11 fps

f. the design accident and scram reactivity shape function, and

g. the moderator temperature at which criticality occurs

In most cases the worth of insequence rods or rod segments in conjunction with the actual values of the other important accident analysis parameters described above, would most likely result in a peak fuel enthalpy substantially less than 280 cal/g design limit.

Should a control drop accident result in a peak fuel energy content of 280 cal/g, fewer than 660 (7 x 7) fuel rods are conservatively estimated to perforate. This would result in an offsite dose well below the guideline value of 10 CFR 100. For 8 x 8 fuel, fewer than 850 rods are conservatively estimated to perforate, with nearly the same consequences as for the 7 x 7 fuel case because of the rod power differences.

The rod worth minimizer provides automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences (reference SAR Section 7.9). It serves as a backup to procedural control of control rod worth. In the event that the rod worth minimizer is out of service when required, a licensed operator or other qualified technical employee can manually fulfill the control rod pattern conformance function of the rod worth minimizer. In this case, the normal procedural controls are backed up by independent procedural controls to assure conformance.

4. The source range monitor (SRM) system performs no automatic safety system function, i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. This is needed for knowledgeable and efficient reactor startup at low neutron levels. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's is provided as an added conservatism.

5. The rod block monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator, who withdraws control rods according to a written sequence. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists. During reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPR's less than the MCPR fuel cladding integrity safety limit. During use of such patterns it is judged that testing of the RBM system to assure its operability prior to withdrawal of such rods will assure that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns.

6. The operability of the Scram Discharge Volume vent and drain valves assures the proper venting and draining of the Volume, so that water accumulation in the Volume does not occur. These specifications provide for the periodic verification that the valves are open, and for the testing of these valves under reactor scram conditions during each Refueling Outage.