3.2/4.2 PROTECTIVE INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

Applicability:

Applies to the plant instrumentation which performs a protective function.

Objective:

To assure the operability of protective instrumentation.

SURVEILLANCE REQUIREMENTS

Applicability:

Applies to the surveillance requirements of the instrumentation that performs a protective function.

Objective:

To specify the type and frequency of surveillance to be applied to protective instrumentation.

SPECIFICATIONS

A. Primary Containment Isolation Functions

When primary containment integrity is required, the limiting conditions of operation for the instrumentation that initiates primary containment isolation are given in Table 3.2-1.

B. Core and Containment Cooling Systems - Initiation and Control

The limiting conditions for operation for the instrumentation that initiates or controls the core and containment cooling systems are given in Table 3.2-2. This instrumentation must be operable when the system(s) it initiates or controls are required to be operable as specified in Specification 3.5.

C. Control Rod Block Actuation

1. The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2-3.

2. a. When a Limiting Control Rod Pattern exists one of the Rod Block Monitors may be bypassed for maintenance and/or testing provided that this condition does not last longer than 24 hours in a 30 day period. If this condition lasts longer than 24 hours in a 30 day period the system shall be tripped.

   The time spent while in a Limiting Control Rod Pattern with one or more Rod Block Monitors bypassed or inoperable and rod withdrawal blocked does not count against the 24 hours in a 30 day period.

   b. One channel may be bypassed above 30% power without a time restriction provided that a Limiting Control Rod Pattern does not exist and the remaining Rod Block Monitor channel is operable.

   c. Both Rod Block Monitor Channels are automatically bypassed at less than 30% rated thermal power or if the selected control rod has one or more adjacent fuel bundles comprising the outer boundary of the reactor core.

A. Primary Containment Isolation Functions

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2-1.

B. Core and Containment Cooling Systems - Initiation and Control

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2-1.

C. Control Rod Block Actuation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2-1.

Venturi tubes are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus only Group 1 valves are closed. For the worst-case accident, main steamline break outside the drywell, this trip setting of 140% of rated steam flow, in conjunction with the flow limiters and main steamline valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperatures remain less than 1500° F, and release of radioactivity to the environs is well below 10 CFR 100 guidelines (reference SAR Sections 14.2.3.9 and 14.2.3.10).

Temperature monitoring instrumentation is provided in the main steamline tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of 200° F is low enough to detect leaks of the order of 5 to 10 gpm; thus it is capable of covering the entire spectrum of breaks. For large breaks, it is a backup to high-steam flow instrumentation discussed above, and for small breaks with the resulting small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

High radiation monitors in the main steamline tunnel have been provided to detect gross fuel failure. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 7 times normal background and main steamline isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident (reference SAR Section 12.2.1.7).

Pressure instrumentation is provided which trips when main steamline pressure drops below 825 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the Refuel and Startup/Hot Standby modes this trip function is bypassed. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valve to open. With the trip set at 825 psig, inventory loss is limited so that fuel is not uncovered and peak cladding temperatures are much less than 1500° F; thus, there are no fission products available for release other than those in the reactor water (reference SAR Section 11.2.3).

The RCIC and the HPCI high flow and temperature instrumentation are provided to detect a break in their respective piping. Tripping of this instrumentation results in actuation of the RCIC or of HPCI isolation valves. Tripping logic for this function is the same as that for the main steamline isolation valves, thus all sensors are required to be operable or in a tripped condition to meet single-failure criteria. The trip settings of 200°F and 300% of design flow and valve closure time are such that core uncovery is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a one-out-of-two taken twice logic circuit. Unlike the reactor scram circuits, however, there is one trip system associated with each function rather than the two trip systems in the reactor protection system. The single-failure criteria are met by virtue of the fact that redundant core cooling functions are provided, e.g., sprays and automatic blowdown and high-pressure coolant injection. The specification requires that if a trip system becomes inoperable, the system which it activates is declared inoperable. For example, if the trip system for core spray A becomes inoperable, core spray A is declared inoperable and the out-of-service specifications of Specification 3.5 govern. This specification preserves the effectiveness of the system with respect to the single-failure criteria even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not go below the MCPR Fuel Cladding Integrity Safety Limit. The trip logic for this function is one out of n; e.g., any trip on one of the six APRM's, eight IRM's, four SRM's will result in a rod block. The minimum instrument channel requirements assure sufficient instrumentation to assure that the single-failure criteria are met. The minimum instrument channel requirements for the RBM may be reduced by one for a short period of time while in a Limiting Control Rod Pattern to allow for maintenance, testing, or calibration. This time period is only ~ 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal. In addition, while the unit is operating in a limiting control pattern with one or more Rod Block Monitors bypassed and control rod withdrawl blocked, this time does not count towards the 24 hours in a 30 day restriction. This time restriction is placed on the Rod Block monitor system to decrease the probability of a Rod Withdrawl Error while in a limiting control rod pattern. With control rod withdrawl blocked all rod withdrawl is prevented, hence the RBM is not required to function.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection, i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

In the refuel and startup/hot standby modes, the APRM rod block function is set at 12% of rated power. This control rod block provides the same type of protection in the Refuel and Startup/Hot Standby modes as the APRM flow-biased rod block does in the Run mode, i.e., prevents control rod withdrawal before a scram is reached.

The RBM rod block function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the fuel cladding integrity safety limit.

Below 30% power, the worst-case withdrawal of a single control rod without rod block action will not violate the fuel cladding integrity safety limit. Thus the RBM rod block function is not required below this power level. If a control rod is selected that has one or more adjacent fuel bundles comprising the outer boundary of the reactor core, the neutron leakage is sufficiently high such that withdrawal of this rod will not violate the fuel cladding integrity Safety limit. Thus the RBM function is not required for control rods that have one or more adjacent fuel bundles comprising the outer boundaries of the core.

The IRM block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of 10 above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity safety limit.

A downscale indication on an APRM is an indication the instrument has failed or is not sensitive enough. In either case the instrument will not respond to changes in control rod motion, and the control rod motion is thus prevented. The downscale trips are set at 3/125 of full scale.

The SRM rod block with ≤ 100 CPS and the detector not full inserted assures that the SRM's are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram discharge volume high water level block provide annunciation for operator action. The alarm setpoint has been selected to provide adequate time to allow determination of the cause of level increase and corrective action prior to automatic scram initiation.

For effective emergency core cooling for small pipe breaks the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met (reference SAR Section 6.2.6.3). The specification preserves the effectiveness of the system during periods of maintenance, testing or calibration and also minimizes the risk of inadvertent operation, i.e., only one instrument channel out of service.

Two radiation monitors are provided on the refueling floor which initiate isolation of the reactor building and operation of the standby gas treatment systems. The trip logic is one out of two. Trip settings of 100 mR/hr for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation

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]	$\leq[0.58W_D + 50]$ $\frac{FRP}{MFLPD}$ [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]	$\leq 0.65W_D + 43$[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 centerline
3	IRM detector not in Startup position [8]	≥2 feet below core centerline
2[5] [6]	SRM upscale	$\leq 10^5$ counts/sec
2[5]	SRM downscale [9]	$\geq 10^2$ counts/sec
1 (per bank)	High water level in scram discharge volume (SDV)	≤ 25 gallons (per bank)
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 and downscale trips need not be operable at less than 30% rated thermal power, or at any power level if the selected control rod has one or more adjacent fuel bundles comprising the outer boundary of the reactor core. The RBM is automatically bypassed at less than 30% rated thermal power or if the selected control rod has one or more adjacent fuel bundles comprising the outer boundary of the reactor core. 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. WD is the percent of drive flow required to produce a rated core flow of 90 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 GT/E 100CPS.

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 IRN 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.

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. Except as provided by Specification 3.2.C.1 and 3.2.C.2 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 and at least once per 92 days, each valve shall be cycled through at least one complete cycle of full travel. At least once each Refueling Outage, the scram discharge volume vent and drain valves will be demonstrated to:

   a. Close within 30 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 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 operations. During reactor operations with certain limiting control rod patterns, the worst-case withdrawal of a single control rod could result in one or more fuel rods with MCPR's less than the MCPR fuel cladding integrity safety limit. During a Limiting control rod pattern, testing of the RBM system will assure its operability prior to withdrawal of such control rods. To facilitate testing while in a limiting control rod pattern one RBM may be bypassed, for brief periods of time to perform maintenance and/or testing without decreasing the reliability of the system, provided the other RBM is operable. Two RBM channels are provided. Tripping one operable channel will block erroneous rod withdrawal soon enough to prevent violation of the MCPR Safety limit. It is the responsibility of the nuclear engineer to identify these limiting control rod 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 pattern.

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.

3.2/4.2 PROTECTIVE INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

Applicability:

Applies to the plant instrumentation which performs a protective function.

Objective:

To assure the operability of protective instrumentation.

SURVEILLANCE REQUIREMENTS

Applicability:

Applies to the surveillance requirements of the instrumentation that performs a protective function.

Objective:

To specify the type and frequency of surveillance to be applied to protective instrumentation.

SPECIFICATIONS

A. Primary Containment Isolation Functions

When primary containment integrity is required, the limiting conditions of operation for the instrumentation that initiates primary containment isolation are given in Table 3.2-1.

A. Primary Containment Isolation Functions

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2-1.

B. Core and Containment Cooling Systems - Initiation and Control

The limiting conditions for operation for the instrumentation that initiates or controls the core and containment cooling systems are given in Table 3.2-2. This instrumentation must be operable when the system(s) it initiates or controls are required to be operable as specified in Specification 3.5.

B. Core and Containment Cooling Systems - Initiation and Control

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2-1.

C. Control Rod Block Actuation

1. The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2-3.

2. a. When a Limiting Control Rod Pattern exists one of the Rod Block Monitors may be bypassed for maintenance and/or testing provided that this condition does not last longer than 24 hours in a 30 day period. If this condition lasts longer than 24 hours in a 30 day period the system shall be tripped.

   The time spent while in a Limiting Control Rod Pattern with one or more Rod Block Monitors bypassed or inoperable and rod withdrawal blocked does not count against the 24 hours in a 30 day period.

   b. One channel may be bypassed above 30% power without a time restriction provided that a Limiting Control Rod Pattern does not exist and the remaining Rod Block Monitor channel is operable.

   c. Both Rod Block Monitor Channels are automatically bypassed at less than 30% rated thermal power or if the selected control rod has one or more adjacent fuel bundles comprising the outer boundary of the reactor core.

C. Control Rod Block Actuation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2-1.

Venturi tubes are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus only Group 1 valves are closed. For the worst-case accident, main steamline break outside the drywell, this trip setting of 140% of rated steam flow, in conjunction with the flow limiters and main steamline valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperatures remain less than 1500° F, and release of radioactivity to the environs is well below 10 CFR 100 guidelines (reference SAR Sections 14.2.3.9 and 14.2.3.10).

Temperature-monitoring instrumentation is provided in the main steamline tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of 200° F is low enough to detect leaks of the order of 5 to 10 gpm; thus it is capable of covering the entire spectrum of breaks. For large breaks, it is a backup to high-steam flow instrumentation discussed above, and for small breaks with the resulting small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

High-radiation monitors in the main steamline tunnel have been provided to detect gross fuel failure. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 7 times normal background and main steamline isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident (reference SAR Section 12.2.1.7).

Pressure instrumentation is provided which trips when main steamline pressure drops below 825 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the Refuel and Startup/Hot Standby modes this trip function is bypassed. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valve to open. With the trip set at 825 psig, inventory loss is limited so that fuel is not uncovered and peak cladding temperatures are much less than 1500° F; thus, there are no fission products available for release other than those in the reactor water (reference SAR Section 11.2.3).

The RCIC and the HPCI high flow and temperature instrumentation are provided to detect a break in their respective piping. Tripping of this instrumentation results in actuation of the RCIC or of HPCI isolation valves. Tripping logic for this function is the same as that for the main steamline isolation valves, thus all sensors are required to be operable or in a tripped condition to meet single-failure criteria. The trip settings of 200°F and 300% of design flow and valve closure time are such that core uncovery is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a one-out-of-two taken twice logic circuit. Unlike the reactor scram circuits, however, there is one trip system associated with each function rather than the two trip systems in the reactor protection system. The single-failure criteria are met by virtue of the fact that redundant core cooling functions are provided, e.g., sprays and automatic blowdown and high-pressure coolant injection. The specification requires that if a trip system becomes inoperable, the system which it activates is declared inoperable. For example, if the trip system for core spray A becomes inoperable, core spray A is declared inoperable and the out-of-service specifications of Specification 3.5 govern. This specification preserves the effectiveness of the system with respect to the single-failure criteria even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not go below the MCPR Fuel Cladding Integrity Safety Limit. The trip logic for this function is one out of n: e.g., any trip on one of the six APRM's, eight IRM's, four SRM's will result in a rod block. The minimum instrument channel requirements assure sufficient instrumentation to assure that the single-failure criteria are met. The minimum instrument channel requirements for the RBM may be reduced by one for a short period of time while in a Limiting Control Rod Pattern to allow for maintenance, testing, or calibration. This time period is only ~ 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal. In addition, while the unit is operating in a limiting control pattern with one or more Rod Block Monitors bypassed and control rod withdrawl blocked, this time does not count towards the 24 hours in a 30 day restriction. This time restriction is placed on the Rod Block monitor system to decrease the probability of a Rod Withdrawl Error while in a limiting control rod pattern. With control rod withdrawl blocked all rod withdrawl is prevented, hence the RBM is not required to function.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection, i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

In the refuel and startup/hot standby modes, the APRM rod block function is set at 12% of rated power. This control rod block provides the same type of protection in the Refuel and Startup/Hot Standby modes as the APRM flow-biased rod block does in the Run mode, i.e., prevents control rod withdrawal before a scram is reached.

The RBM rod block function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the fuel cladding integrity safety limit.

Below 30% power, the worst-case withdrawal of a single control rod without rod block action will not violate the fuel cladding integrity safety limit. Thus the RBM rod block function is not required below this power level. If a control rod is selected that has one or more adjacent fuel bundles comprising the outer boundary of the reactor core, the neutron leakage is sufficiently high such that withdrawal of this rod will not violate the fuel cladding integrity Safety limit. Thus the RBM function is not required for control rods that have one or more adjacent fuel bundles comprising the outer boundaries of the core.

The IRM block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of 10 above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity safety limit.

A downscale indication on an APRM is an indication the instrument has failed or is not sensitive enough. In either case the instrument will not respond to changes in control rod motion, and the control rod motion is thus prevented. The downscale trips are set at 3/125 of full scale.

The SRM rod block with ≤ 100 CPS and the detector not full inserted assures that the SRM's are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram discharge volume high water level block provide annunciation for operator action. The alarm setpoint has been selected to provide adequate time to allow determination of the cause of level increase and corrective action prior to automatic scram initiation.

For effective emergency core cooling for small pipe breaks the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met (reference SAR Section 6.2.6.3). The specification preserves the effectiveness of the system during periods of maintenance, testing or calibration and also minimizes the risk of inadvertent operation, i.e., only one instrument channel out of service.

Two radiation monitors are provided on the refueling floor which initiate isolation of the reactor building and operation of the standby gas treatment systems. The trip logic is one out of two. Trip settings of 100 mR/hr for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation

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)	$\leq[0.58W_D + 50] \frac{FRP}{MFLPD}$ (2)
2	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)	$\leq 0.65W_D + 42$(2)
1	Rod block monitor downscale(7)	$\geq$3/125 full scale
3	IRM downscale(3) (8)	$\geq$3/125 full scale
3	IRM upscale(8)	$\leq$108/125 full scale
2(5)	SRM detector not in Startup position (4)	$\geq$2 feet below core centerline
3	IRM detector not in Startup position (8)	$\geq$2 feet below core centerline
2(5) (6)	SRM upscale	$\leq 10^5$ counts/sec
2(5)	SRM downscale (9)	$\geq 10^2$ counts/sec
1 (per bank)	High water level in scram discharge volume (SDV)	$\leq$ 25 gallons (per bank)
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 and downscale trips need not be operable at less than 30% rated thermal power, or at any power level if the selected control rod has one or more adjacent fuel bundles comprising the outer boundary of the reactor core. The RBM is automatically bypassed at less than 30% rated thermal power or if the selected control rod has one or more adjacent fuel bundles comprising the outer boundary of the reactor core. 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.

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. Except as provided by Specification 3.2.C.1 and 3.2.C.2 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 on line 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 and at least once per 92 days, each valve shall be cycled through at least one complete cycle of full travel. At least once each Refueling Outage, the scram discharge volume vent and drain valves will be demonstrated to:

   a. Close within 30 seconds after receipt of a signal for control rods to scram, and

   b. Open when the scram signal is reset.

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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 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 erronenus rod withdrawal from locations of high power density during high power operations. During reactor operations with certain limiting control rod patterns, the worst-case withdrawal of a single control rod could result in one or more fuel rods with MCPR's less than the MCPR fuel cladding integrity safety limit. During a Limiting control rod pattern, testing of the RBM system will assure its operability prior to withdrawal of such control rods. To facilitate testing while in a limiting control rod pattern one RBM may be bypassed, for brief periods of time to perform maintenance and/or testing without decreasing the reliability of the system, provided the other RBM is operable. Two RBM channels are provided. Tripping one operable channel will block erroneous rod withdrawal soon enough to prevent violation of the MCPR Safety limit. It is the responsibility of the nuclear engineer to identify these limiting control rod 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 pattern.

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 2

SUMMARY OF CHANGES

A total of twenty four (24) changes to the Quad Cities Station Units 1 and 2 Technical Specifications have been identified (12 per unit) and are listed below as follows:

1. Page 3.2/4.2-1, DPR-29 and 30

   (a) Limiting Condition Operation (LCO) - Delete Item C.2 in entirety and replace with new items 2.a., 2.b., and 2.c..

   This change is strictly a numbering change and is administratively in nature.

   (b) LCO, Technical Specification 3.2.C. - Create new Item 3.2.C.2.a., which reads as follows: "When a Limiting Control Rod Pattern exists one or more...does not count against the 24 hours in a 30 day period."

   This section was changed to clarify time restrictions on Rod Block Monitor bypassing. It now states that while in a limiting control pattern, one Rod Block Monitor may be bypassed for maintenance/ testing for no longer than 24 hours in a 30 day period; unless in this condition rod withdrawal is blocked, then this time does not count against the 24 hrs. in a 30 day period.

   (c) LCO, Technical Specification 3.2.C. - Create new Item 3.2.C.2.b., which reads as follows: "One channel may be bypassed above 30% power...and the remaining Rod Block Monitor Channel is operable."

   This section was changed so that if a limiting control rod pattern does not exist, then one Rod Block Monitor may be bypassed for any length of time, provided the other Rod Block Monitor is operable. Rod Withdrawal need be blocked in this case.

   (d) LCO, Technical Specification 3.2.C. - Create new Item 3.2.C.2.c., which reads as follows: "Both Rod Block Monitor Channels are automatically bypassed at less...of the reactor core."

   This section would permit both Rod Block Monitors to be automatically bypassed below 30% rated thermal power and on edge control rods.

2. Page 3.2/4.2-6, DPR-29 and 30

   (a) Delete last paragraph of Bases.

   This paragraph to the Bases is being dropped and replaced with a new section. The new section encompasses the changes which result from the proposed clarifications to the Rod Block Monitor Technical Specifications.

3. Page 3.2/4.2-6a, DPR-29 and 30

(a) Create new paragraph which reads as follows: "The Control Rod Block functions are provided to prevent...hence the RBM is not required to function."

This new section provides clarification that while in a limiting control rod pattern, one Rod Block Monitor may be bypassed for a short period of time to perform maintenance and/or testing. This provision is currently in the Bases, however, would otherwise have been deleted based on the change described in Item 2(a). In addition, while the unit is operating in a limiting control pattern with one or more Rod Block Monitors bypassed and control rod withdrawal block, this time does not count towards the 24 hours in a 30 day restriction.

4. Page 3.2/4.2-7, DPR-29 and 30

(a) Bases - fourth paragraph, third line. Following the sentence which ends with the words "power level.", insert the following: "If a control rod is selected that has one or more adjacent fuel bundles...the outer boundaries of the core".

This addition to the fourth paragraph clarifies that both Rod Block Monitors are automatically bypassed below 30% power and when a control rod, with one or more fuel bundles residing on the reactors perifery, is selected.

5. Page 3.2/4.2-14, DPR-29 and 30

(a) Note 1, fourth line - Following the words "RBM upscale", insert the words "and downscale trips", so that the sentence now reads "The RBM upscale and downscale trips need be operable at less than 30% rated thermal power,"

(b) Note 1 (continued) - Following the words "at less than 30% rated thermal power," insert the words "or at any power level if the selected control rod...comprising the outer boundary of the reactor core" so that the sentence now reads, "The RBM upscale and downscale trips...or at any power level...of the reactor core."

(c) Note 1 (continued) - Delete the sentence "One channel may be bypassed above 30% rated thermal power provided that a limiting control rod pattern does not exist.

These changes (5a, 5b, and 5c), provide clarification as to when the upscale and downscale trips of the Rod Block Monitor are not required and when they are automatically bypassed.

6. Page 3.3/4.3-3, DPR-29 and 30

   (a) LCO, Technical Specification 3.3.B.5 - Insert the words "Except as provided by Specification 3.2.C.1 and 3.2.C.2" so the sentence now reads "except as provided by the by Specification 3.2.C.1 and 3.2.C.2 during operation with a limiting control rod patterns..."

   This addition clarifies that while in a limiting control rod pattern, one Rod Block Monitor may be bypassed for maintenance and/or testing. In addition, it also clarifies that an edge rod may have both Rod Block Monitors bypassed while in a Limiting Control Rod pattern.

7. Page 3.3/4.3-9, DPR-29 and 30

   (a) Bases, Item 5, third line - Delete the section beginning with the words, "Two channels are provided..." through the sentence containing the words "identify these limiting patterns". Replace with section that begins with words "During reactor operations with certain limiting control rod patterns,..." through the sentence that contains the words "identify these limiting control rod patterns".

   This addition clarifies the Rod Block Monitor operability requirement during a limiting control rod pattern.

ATTACHMENT 3

BASIS FOR SIGNIFICANT HAZARDS CONSIDERATION

CLARIFICATION OF ROD BLOCK MONITOR (RPM)

OPERABILITY AND BYPASS TIME REQUIREMENTS

The proposed Technical Specification amendments to sections 3.2.C, Table 3.2-3, 3.3.B.5 and the Basis for these sections, are being submitted to clarify interpretations of these sections. This Technical Specification amendment will clarify bypass time limitations on the Rod Block Monitors and also Rod Block Monitor operability.

The proposed Technical Specification amendment allows 1 Rod Block Monitor to be bypassed without any time limitations provided the other Rod Block Monitor is operable and a Limiting Control Rod Pattern does not exist. One Rod Block Monitor is capable of preventing the worst case unrestricted rod withdrawl from violating the MCPR Safety Limit. By definition a Limiting Control Rod Pattern is a condition in which the worst case, unrestricted withdrawl of a control rod could violate the MCPR Safety Limit. However, due to changing conditions with core flow, xenon, or control rod movement it is difficult to determine when the unit enters a limiting control rod pattern. Therefore providing for one Rod Block Monitor to be operable while not in a limiting control rod pattern will prevent the worst case unrestricted rod withdrawal if a limiting control rod pattern is entered.

While the unit is operating in a Limiting Control rod pattern both Rod Block monitors must be operable, except for maintenance and/or testing in which case one may be bypassed for 24 hours in 30 days. This short period of time will allow proper testing of the Rod Block Monitors and will not significantly increase the risk of the worst case unrestricted rod withdrawal to violate the MCPR Safety Limit.

In addition, while the unit is operating in a limiting control pattern with one or more Rod Block Monitors bypassed and control rod withdrawal block, this time does not count towards the 24 hours in a 30 day restriction. This time restriction is placed on the Rod Block monitor system to decrease the probability of a Rod Withdrawal Error while in a limiting control rod pattern. With control rod withdrawal blocked all rod withdrawal is prevented, hence the RBM is not required to function. The 24 hour time should only apply when the RBM is required to provide a Rod Block Function.

When reactor power is less than 30% rated core thermal power, or an edge rod is selected, at any power, both rod block monitors are automatically bypassed. General Electric's Equipment Manual APED-5706, November 1968, documents that below this power level the bundle powers are low enough that withdrawal of the strongest rod will not violate the MCPR Safety Limit. Analysis also showed significant neutron leakage on the edge fuel assemblies of the core such that withdrawal of any edge control rod will not violate the MCPR Safety Limit.

These changes have been reviewed by Commonwealth Edison and we believe they do not present a Significant Hazards Consideration. The basis for our determination is documented as follows:

BASIS FOR NO SIGNIFICANT HAZARD DETERMINATION

Commonwealth Edison Company has evaluated this proposed amendment and determined that it involves no significant hazards considerations. In accordance with the criteria of 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated because:

   (a) The proposed Technical Specification changes require at least one Rod Block Monitor to be operable whenever the worst case unrestricted rod withdrawal error could violate the MCPR Safety Limit. One Rod Block monitor is sufficient to prevent the MCPR Safety Limit from being violated during the worst case, unrestricted rod withdrawl. Therefore, this does not involve a significant increase in the probability or consequence of an accident previously evaluated.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated; because;

   (a) A review of the proposed Technical Specification changes does not reveal a new or different kind of accident from any previously evaluated. This proposed amendment does not change the times the Rod Block monitor is needed and therefore does not create the possibility of a new or different kind of accident than previously was evaluated.

3) Involve a significant reduction in the margin of safety, because;

(a) The proposed Technical Specification change requires at least one Rod Block Monitor to be operable when reactor power is sufficient so that the worst case unrestricted rod withdrawl could violate the MCPR Safety Limit. The only time that the worst case unrestricted rod withdrawl could violate the MCPR Safety Limit is when the unit is operating in a Limiting Control rod pattern. However, due to changing conditions with core flow, Xenon, or control rod movement it is difficult to determine when the unit enters a limiting control rod pattern. Therefore, providing one Rod Block Monitor to be operable while not in a limiting control rod pattern will prevent the worst case unrestricted rod withdrawl if a limiting control rod pattern is entered. In addition when a limiting control rod pattern exists both Rod Block Monitors must be operable or rod withdrawl be blocked, except for maintenance and/or testing for brief periods of time. This ensures with a high degree of certainty that the worst case rod withdrawl error will not violate the MCPR Safety Limit. Hence, the changes do not result in a decrease in the margin of safety.

Therefore since the proposed license amendment satisfies the criteria specified in 10 CFR 50.92, Commonwealth Edison has determined that a no significant hazards consideration exists for this license amendment. We request its approval in accordance with the provisions of 10 CFR 50.91(a)(4).