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DEC 23 1982

Docket File
NRC PDR
Local PDR
ORB#2 Rdg.
D. Eisenhut
S. Norris
R. Bevan
OELD
SECY
OI&E - LJHarmon-2
T. Barnhart - 8
L. Schneider
D. Brinkman
ACRS 10
OPA Clare Miles
R. Diggs
NSIC
Gray
ASLAB
E. Jordan, DEQA
J. Taylor, DRP

XTRA-5
KEccleston

Docket Nos. 50-254
50-265

DEC 23 1982

Mr. L. DelGeorge
Director of Nuclear Licensing
Commonwealth Edison Company
P. O. Box 767
Chicago, Illinois 60690

Dear Mr. DelGeorge:

The Commission has issued the enclosed Amendment Nos. 84 and 77 to Facility Operating License Nos. DPR-29 and DPR-30 for the Quad Cities Nuclear Power Station Units 1 and 2. These amendments consist of changes to the Technical Specifications as proposed in your application dated October 14, 1980 and supplemented by your letter dated October 22, 1981.

The amendments expand the Technical Specifications for the scram discharge volume (SDV) to include surviellance requirements for SDV vent and drain valves and limiting conditions for operation (LCO) and surviellance requirements for the reactor protection system (RPS) and control rod block SDV limit switches.

Copies of our Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

"ORIGINAL SIGNED BY:"

Roby B. Bevan, Project Manager
Operating Reactors Branch No. 2
Division of Licensing

Enclosures:
1. Amendment No. 84 to DPR-29
2. Amendment No. 77 to DPR-30
3. Safety Evaluation
4. Notice

cc w/enclosure
See next page

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DATE	12/8/82	12/8/82	12/9/82	12/10/82	12/14/82		

Mr. L. DelGeorge
Commonwealth Edison Company

cc:

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The Honorable Tom Corcoran
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Washington, D.C. 20515

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY
AND
IOWA ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 84
License No. DPR-29

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated October 14, 1980, as supplemented October 22, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility License No. DPR-29 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 84, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 23, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 84

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

Revise the Appendix "A" Technical Specifications as follows:

Remove	Insert
3.2/4.2-14	3.2/4.2-14
3.2/4.2-16	3.2/4.2-16
3.3/4.3-3	3.3/4.3-3
3.3/4.3-9	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	≤10⁵ counts/sec
2[5]	SRM downscale[9]	≥10² 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 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 patter 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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY
AND
IOWA ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES STATION UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 77
License No. DPR-30

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated October 14,1980, as supplemented October 22, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 DFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility License No. DPR-30 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.77 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

D B Vassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 23, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 77

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Revise the Appendix "A" Technical Specifications as follows:

Remove	Insert
3.2/4.2-14	3.2/4.2-14
3.2/4.2-16	3.2/4.2-16
3.3/4.3-3	3.3/4.3-3
3.3/4.3-9	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]	$\leq 0.650W + 42^{(2)}$
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

Amendment No. 77, 77

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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NO. DPR-29

AND

AMENDMENT NO. 77 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES NUCLEAR POWER STATION UNITS 1 AND 2

DOCKET NOS. 50-254 AND 50-265

Introduction

As a result of events involving common cause failures of SDV limit switches and SDV drain valve operability, the NRC staff issued IE Bulletin 80-14 on June 12, 1980. In addition, the staff sent a letter dated July 7, 1980 to all operating BWR licensees requesting that they propose Technical Specification changes to provide surveillance requirements for SDV vent and drain valves and LCO/surveillance requirements on SDV limit switches. Model Technical Specifications were enclosed with this letter to provide guidance to licensees for preparation of the requested submittals. By letter dated October 14, 1980, as supplemented October 22, 1981, Commonwealth Edison Company (the licensee) submitted the proposed changes to the Technical Specifications.

Evaluation

The enclosed report (TER-C5506-63/65) was prepared for us by Franklin Research Center (FRC) as part of a technical assistance contract program. The FRC report provides its technical evaluation of the compliance of the licensee's submittal with NRC provided criteria.

FRC has concluded that the licensee's response does not meet the explicit requirements of paragraph 3.3-6 and Table 3.3.6-1 of the NRC staff's Model Technical Specifications. However, the FRC report concludes that technical bases are defined on p. 50 of the staff's "Generic Safety Evaluation Report BWR Scram Discharge System", dated December 1, 1980 that permit consideration of this departure from the explicit requirements of the Model Technical Specifications. We conclude that these technical bases justify a deviation from the explicit requirements of the Model Technical Specifications.

In addition, FRC has also concluded that the proposed Quad Cities Units 1 and 2 Technical Specifications do not meet the Model Technical Specification requirements of paragraphs 4.3.1.1 and Table 4.3.1.1-1 for SDV water level high channel functional test requirements. However, the FRC TER concludes that the proposed surveillance requirements for SDV water level high are acceptable, since the licensee is installing a second instrument volume at each unit and the licensee is providing four reactor protection system level instruments for each of the two instrument volumes, for a total of eight instruments for the RPS. The Model Technical Specifications were developed for plants which have only one instrument volume (four RPS level switches); therefore, the second instrument volume significantly improves the design and reliability of the SDV. Taking this into account, we conclude that the technical bases justify a deviation from the explicit requirements of the Model Technical Specifications.

FRC has concluded that the licensee's proposed Technical Specification revisions meet our criteria without the need for further revision.

Based upon our review of the contractor's report of its evaluations, we conclude that the licensee's proposed Technical Specifications satisfy our requirements for surveillance of SDV vent and drain valves and for LCOs and surveillance requirements for SDV limit instrumentation. Consequently, we find the licensee's proposed Technical Specifications acceptable.

Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: December 23, 1982

Enclosure: TER

Principal Contributors: K. T. Eccleston, R. B. Bevan

TECHNICAL EVALUATION REPORT

BWR SCRAM DISCHARGE VOLUME LONG-TERM MODIFICATIONS

COMMONWEALTH EDISON COMPANY

QUAD CITIES STATION UNITS 1 AND 2

NRC DOCKET NO. 50-254, 50-265

NRC TAC NO. 42224, 42225

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 2

FRC TASKS 63, 65

Prepared by

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July 13, 1982

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TECHNICAL EVALUATION REPORT

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CONTENTS

FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

SUMMARY

This technical evaluation report reviews and evaluates proposed Phase 1 changes in the Quad Cities Station Units 1 and 2 Technical Specifications for scram discharge volume (SDV) long-term modifications regarding surveillance requirements for SDV vent and drain valves and the limiting condition for operation (LCO)/surveillance requirements for reactor protection system and control rod withdrawal block SDV limit switches. Conclusions were based on the degree of compliance of the Licensee's submittal with criteria from the Nuclear Regulatory Commission (NRC) staff's Model Technical Specifications.

The revised page 3.3/4.3-3, with the Licensee's agreement to incorporate a revision into the proposed specifications changes that requires cycling each valve at least one complete cycle of full travel at least quarterly, complies with the NRC staff's Model Technical Specifications requirements of paragraphs 4.1.3.1.1a and 4.1.3.1.1b.

The original pages 3.1/4.1-12 and 3.1/4.1-13, Table 4.1.1, of the Quad Cities Station Units 1 and 2 Technical Specifications, which provide for the reactor protection system SDV limit switches water level-high Channel Functional Test to be performed once per 3 months, do not meet the surveillance requirement (paragraph 4.3.1.1, Table 4.3.1.1-1, of the NRC Staff's Model Technical Specifications) for the test to be performed monthly. However, the Licensee is installing a second instrument volume containing four additional limit switches, for a total of eight limit switches, for the reactor protection system. This increases significantly the reliability of the system and provides technical bases for acceptance of the proposed surveillance requirements to perform the Channel Functional Test quarterly.

To meet the NRC staff's Model Technical Specifications requirements of paragraph 4.3.1.1 and Table 4.3.1.1-1, the Licensee agreed to revise page 3.1/4.1-14 of the present Quad Cities Station Units 1 and 2 Technical Specifications to incorporate in Table 4.1-2 the Calibration Test "each refueling" for "Instrument Channel - SDV Water Level High." The Calibration

Test will consist of physical inspection and actuation of the level switches using water columns.

The NRC staff's Model Technical Specifications surveillance requirements in paragraph 3.3.6, Table 3.3.6-1, paragraph 4.3.6, and Table 4.3.6-1 for control rod block SDV scram trip bypassed are not applicable to the operational conditions of startup, run, and refuel with more than one control rod withdrawn. Therefore, the Licensee agreed to delete "Instrument Channel - SDV High Water Level Scram Trip Bypassed" from the proposed revision of page 3.2/4.2-14, Table 3.2-3, and page 3.2/4.2-16, Table 4.2-1.

The existing SDV system has only one trip system with one instrument channel containing one control rod withdrawal block SDV limit switch and is acceptable. To reflect this, the Licensee agreed to revise the first sentence of Note 1 in Table 3.2.3 on original page 3.2/4.2-14.

To meet the NRC staff's Model Technical Specifications requirements of paragraph 4.3.6 and Table 4.3.6-1, the Licensee agreed to incorporate into Table 4.2-1 on revised page 3.2/4.2-16 the Calibration Test "each refueling" instead of "Not applicable" for "Instrument Channel-Rod Blocks, High Water Level in Scram Discharge Volume." Channel Calibration with the Magnetrol level switch will consist of physical inspection and actuation of the switch using a water column.

The remaining surveillance requirements are met by revised pages 3.2/4.2-14, 3.2/4.2-16, 3.3/4.3-3, 3.3/4.3-9, and original, unrevised pages 3.1/4.1-12, 3.1/4.1-13, 3.1/4.1-14, and 3.3/4.3-10 of the Quad Cities Station Units 1 and 2 Technical Specifications. Table 5-1 on pages 28 and 29 of this report summarizes the evaluation results.

1. INTRODUCTION

1.1 PURPOSE OF THE TECHNICAL EVALUATION

The purpose of this technical evaluation report (TER) is to review and evaluate the proposed changes in the Technical Specifications of the Quad Cities Station Units 1 and 2 boiling water reactor (BWR) in regard to "BWR Scram Discharge Volume Long Term Modification," specifically:

- o surveillance requirements for scram discharge volume (SDV) vent and drain valves
- o limiting condition for operation (LCO)/surveillance requirements for the reactor protection system
- o LCO/surveillance requirements for the control rod withdrawal block SDV limit switches.

The evaluation used criteria proposed by the NRC staff in Model Technical Specifications (see Appendix A of this report). This effort is directed toward the NRC objective of increasing the reliability of installed BWR scram discharge volume systems, the need for which was made apparent by events described below.

1.2 GENERIC ISSUE BACKGROUND

On June 13, 1979, while the reactor at Hatch Unit 1 was in the refuel mode, two SDV high level switches had been modified, tested, and found inoperable. The remaining switches were operable. Inspection of each inoperable level switch revealed a bent float rod binding against the side of the float chamber.

On October 19, 1979, Brunswick Unit 1 reported that water hammer due to slow closure of the SDV drain valve during a reactor scram damaged several pipe supports on the SDV drain line. Drain valve closure time was approximately 5 minutes because of a faulty solenoid controlling the air supply to the valve. After repair, to avoid probable damage from a scram, the unit was started with the SDV vent and drain valves closed except for periodic draining. During this mode of operation, the reactor scrammed due to a high water level in the

SDV system without prior actuation of either the high level alarm or rod block switch. Inspection revealed that the float ball on the rod block switch was bent, making the switches inoperable. The water hammer was reported to be the cause of these level switch failures.

As a result of these events involving common-cause failures of SDV limit switches and SDV drain valve operability, the NRC issued IE Bulletin 80-14, "Degradation of BWR Scram Discharge Volume Capability," on June 12, 1980 [1]. In addition, to strengthen the provisions of this bulletin and to ensure that the scram system would continue to work during reactor operation, the NRC sent a letter dated July 7, 1980 [2] to all operating BWR licensees requesting that they propose Technical Specifications changes to provide surveillance requirements for reactor protection system and control rod block SDV limit switches. The letter also contained the NRC staff's Model Technical Specifications to be used as a guide by licensees in preparing their submittals.

Meanwhile, during a routine shutdown of the Browns Ferry Unit 3 reactor on June 28, 1980, 76 of 185 control rods failed to insert fully. Full insertion required two additional manual scrams and an automatic scram for a total elapsed time of approximately 15 minutes between the first scram initiation and the complete insertion of all the rods. On July 3, 1980, in response to both this event and the previous events at Hatch Unit 1 and Brunswick Unit 1, the NRC issued (in addition to the earlier IE Bulletin 80-14) IE Bulletin 80-17 followed by five supplements. These initiated short-term and long-term programs described in "Generic Safety Evaluation Report BWR Scram Discharge System," NRC Staff, December 1, 1980 [9] and "Staff Report and Evaluation of Supplement 4 to IE Bulletin 80-17 (Continuous Monitoring Systems)" [10].

Analysis and evaluation of the Browns Ferry Unit 3 and other SDV system events convinced the NRC staff that SDV systems in all BWRs should be modified to assure long-term SDV reliability. Improvements were needed in three major areas: SDV-IV hydraulic coupling, level instrumentation, and system isolation. To achieve these objectives, an Office of Nuclear Reactor Regulation (NRR) task force and a subgroup of the BWR Owners Group developed Revised Scram Discharge

System Design and Safety Criteria for use in establishing acceptable SDV systems modifications [9]. Also, an NRC letter dated October 1, 1980 requested all operating BWR licensees to reevaluate installed SDV systems and modify them as necessary to comply with the revised criteria.

In Reference 9, the SDV-IV hydraulic coupling at the Big Rock Point, Brunswick 1 & 2, Duane Arnold, and Hatch 1 & 2 BWRs was judged acceptable. The remaining BWRs will require modification to meet the revised SDV-IV hydraulic coupling criteria, and all operating BWRs may require modification to meet the revised instrumentation and isolation criteria. The changes in Technical Specifications associated with this effort will be carried out in two phases:

Phase 1 - Improvements in surveillance for vent and drain valves and instrument volume level switches.

Phase 2 - Improvements required as a result of long-term modifications made to comply with revised design and performance criteria.

This TER is a review and evaluation of Technical Specifications changes proposed for Phase 1.

1.3 PLANT-SPECIFIC BACKGROUND

The July 7, 1980 NRC letter [2] not only requested all BWR licensees to amend their facilities' Technical Specifications with respect to control rod drive SDV capability, but enclosed the NRC staff's proposed Model Technical Specifications (see Appendix A of this TER) as a guide for the licensees in preparing the requested submittals and as a source of criteria for a technical evaluation of the submittals. This TER is a review and evaluation of Technical Specifications changes for the Quad Cities Station Units 1 and 2 proposed by the Licensee, Commonwealth Edison (CE), in letters dated October 14, 1980 and October 22, 1981 (see Appendices B and C, respectively) in regard to "BWR Scram Discharge Volume (SDV) Long-Term Modifications" and, specifically, the surveillance requirements for SDV vent and drain valves and the limiting condition for operation (LCO)/surveillance requirements for the reactor protection system and control rod withdrawal block SDV limit switches. The adequacy with which the CE information documented compliance of the proposed Technical Specifications changes with the NRC staff's Model Technical Specifications is also assessed.

2. REVIEW CRITERIA

The criteria established by the NRC staff's Model Technical Specifications involving surveillance requirements of the main SDV components and instrumentation cover three areas of concern:

- o surveillance requirements for SDV vent and drain valves
- o LCO/surveillance requirements for reactor protection system SDV limit switches
- o LCO/surveillance requirements for control rod block SDV limit switches.

2.1 SURVEILLANCE REQUIREMENTS FOR SDV DRAIN AND VENT VALVES

The surveillance criteria of the NRC staff's Model Technical Specification for SDV drain and vent valves are:

"4.1.3.1.1 - The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. Verifying each valve to be open* at least once per 31 days, and
b. Cycling each valve at least one complete cycle of full travel at least once per 92 days.

*These valves may be closed intermittently for testing under administrative controls."

The Model Technical Specifications require testing the drain and vent valves, checking at least once in every 31 days that each valve is fully open during normal operation, and cycling each valve at least one complete cycle of full travel under administrative controls at least once per 92 days.

Full opening of each valve during normal operation indicates that there is no degradation in the control air system and its components that control the air pressure to the pneumatic actuators of the drain and vent valves. Cycling each valve checks whether the valve opens fully and whether its movement is smooth, jerky, or oscillatory.

During normal operation, the drain and vent valves stay in the open position for very long periods. A silt of particulates such as metal chips and flakes, various fibers, lint, sand, and weld slag from the water or air may accumulate at moving parts of the valves and temporarily freeze them. A strong breakout force may be needed to overcome this temporary freeze, producing a violent jerk which may induce a severe water hammer if it occurs during a scram or a scram resetting. Periodic cycling of the drain and vent valves is the best method to clear the effects of particulate silting, thus promoting smooth opening and closing and more reliable valve operation. Also, in case of improper valve operation, cycling can indicate whether excessive pressure transients may be generated during and after a reactor scram which might damage the SDV piping system and cause a loss of system integrity or function.

2.2 LCO/SURVEILLANCE REQUIREMENTS FOR REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES

The paragraphs of the NRC staff's Model Technical Specifications pertinent to LCO/surveillance requirements for reactor protection system SDV limit switches are:

"3.3.1 - As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

Table 3.3.1-1. Reactor Protection System Instrumentation

Functional Unit	Applicable Operational Conditions	Minimum Operable Channels Per Trip System (a)	Action
8. Scram Discharge Volume Water Level-High	1,2,5 (h)	2	4

Table 3.3.1-2. Reactor Protection System Response Times

Functional Unit	Response Time (Seconds)
8. Scram Discharge Volume Water Level-High	NA

"4.3.1.1 - Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

Table 4.3.1.1-1. Reactor Protection System Instrumentation Surveillance Requirements

Functional Unit	Channel Check	Channel Functional Test	Channel Calibration	Operational Conditions in Which Surveillance Required
8. Scram Discharge Volume Water Level-High	NA	M	R	1,2,5

Notation (a) A channel may be placed in an inoperable status up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2

Action 4: In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

*Except movement of IRM, SRM or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2."

Paragraph 3.3.1 and Table 3.3.1-1 of the Model Technical Specifications require the functional unit of SDV water level-high to have at least 2 operable channels containing 2 limit switches per trip system, a total of 4 operable channels containing 4 limit switches per 2 trip systems for the reactor protection system which automatically initiates a scram. The technical objective of these requirements is to provide 1-out-of-2-taken-twice logic for the reactor protection system. The response time of the reactor protection system for the functional unit of SDV water level-high should be measured and kept available (it is not given in Table 3.3.1-2).

Paragraph 4.3.1.1 and Table 4.3.1.1-1 give reactor protection system instrumentation surveillance requirements for the functional unit of SDV water level-high. Each reactor protection system instrumentation channel containing a limit switch should be shown to be operable by the Channel Functional Test monthly and Channel Calibration at each refueling outage.

2.3 LCO/SURVEILLANCE REQUIREMENTS FOR CONTROL ROD WITHDRAWAL BLOCK SDV LIMIT SWITCHES

The NRC staff's Model Technical Specifications specify the following LCO/surveillance requirements for control rod withdrawal block SDV limit switches:

"3.3.6 - The control rod withdrawal block instrumentation channel shown in Table 3.3.6-1 shall be OPERABLE with trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

Table 3.3.6-1. Control Rod Withdrawal Block Instrumentation

Trip Function	Minimum Operable Channels Per Trip Function	Applicable Operational Conditions	Action
5. Scram Discharge Volume			
a. Water level-high	2	1, 2, 5**	62
b. Scram trip bypassed	1	(1, 2, 5**)	62

ACTION 62: With the number of OPERABLE channels less than required by the minimum OPERABLE channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

**With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

Table 3.3.6-2 Control Rod Withdrawal Block Instrumentation Setpoints

Trip Function	Trip Setpoint	Allowable Value
5. Scram Discharge Volume		
a. Water level-high	To be specified	NA
b. Scram trip bypassed	NA	NA"

"4.3.6 - Each of the above control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

Table 4.3.6-1. Control Rod Withdrawal Block Instrumentation Surveillance Requirements

Trip Function	Channel Check	Channel Functional Test	Channel Calibration	Operational Conditions in Which Surveillance Required
5. Scram Discharge Volume				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Trip Bypassed	NA	M	NA	(1, 2, 5**)

**With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2."

Paragraph 3.3.6 and Table 3.3.6-1 of the Model Technical Specifications require the control rod withdrawal block instrumentation to have at least 2 operable channels containing 2 limit switches for SDV water level-high and 1 operable channel containing 1 limit switch for SDV scram trip bypassed. The technical objective of these requirements is to have at least one channel containing one limit switch available to monitor the SDV water level when the other channel with a limit switch is being tested or undergoing maintenance. The trip setpoint for control rod withdrawal block instrumentation monitoring

SDV water level-high should be specified as indicated in Table 3.3.6-2. The trip function prevents further withdrawal of any control rod when the control rod block SDV limit switches indicate water level-high.

Paragraph 4.3.6 and Table 4.3.6-1 require that each control rod withdrawal block instrumentation channel containing a limit switch be shown to be operable by the Channel Functional Test once per 3 months for SDV water level-high, by the Channel Functional Test once per month for SDV scram trip bypassed, and by Channel Calibration at each refueling outage for SDV water level-high.

The Surveillance Criteria of the BWR Owners Subgroup given in Appendix A, "Long-Term Evaluation of Scram Discharge System," of "Generic Safety Evaluation Report BWR Scram Discharge System," written by the NRC staff and issued on December 1, 1980, are:

1. Vent and drain valves shall be periodically tested.
2. Verifying and level detection instrumentation shall be periodically tested in place.
3. The operability of the entire system as an integrated whole shall be demonstrated periodically and during each operating cycle, by demonstrating scram instrument response and valve function at pressure and temperature at approximately 50% control rod density.

Analysis of the above criteria indicates that the NRC staff's Model Technical Specifications requirements, the acceptance criteria for the present TER, fully cover the BWR Owners Subgroup Surveillance Criteria 1 and 2 and partially cover Criterion 3.

3. METHOD OF EVALUATION

The CE submittal for the Quad Cities Station Units 1 and 2 was evaluated in two stages, initial and final.

During the initial evaluation, only the NRC staff's Model Technical Specifications requirements were used to determine if:

- o the Licensee's submittal was responsive to the July 7, 1980 NRC request for proposed Technical Specifications changes involving the surveillance requirements of the SDV vent and drain valves, LCO/surveillance requirements for reactor protection system SDV limit switches, and LCO/surveillance requirements for control rod block SDV limit switches
- o the submitted information was sufficient to permit a detailed technical evaluation.

During the final evaluation, in addition to the NRC staff's Model Technical Specifications requirements, background material in References 1 through 10, pertinent sections of "Commonwealth Edison Quad Cities Station Units 1 and 2 Safety Analysis Report," and Quad Cities Technical Specifications were studied to determine the technical bases for the design of SDV main components and instrumentation. Subsequently, the Licensee's response was compared directly to the requirements of the NRC staff's Model Technical Specifications. The findings of the final evaluation are presented in Section 4 of this report.

The initial evaluation concluded that the Licensee's submittal was responsive to the NRC request of July 7, 1980, but certain information was not available. A Request for Additional Information (RAI) was sent to CE by the NRC on September 2, 1981. Thus, this TER is based on the initial submittal and the Licensee's response dated October 22, 1981 (see Appendix C) to the RAI.

4. TECHNICAL EVALUATION

4.1 SURVEILLANCE REQUIREMENTS FOR SDV DRAIN AND VENT VALVES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 4.1.3.1.1 requires demonstrating that the SDV drain and vent valves are operable by:

a. verifying each valve to be open at least once per 31 days (valves may be closed intermittently for testing under administrative controls)

b. cycling each valve at least one complete cycle of full travel at least once per 92 days.

LICENSEE RESPONSE

The Licensee proposed to revise page 3.3/4.3-3 of the Quad Cities Station Units 1 and 2 Technical Specifications by adding paragraph 6:

"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."

and provided a revision of page 3.3/4.3-9 with this pertinent statement:

"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."

The Licensee's answer to the RAI regarding cycling the drain and vent valves at least one complete cycle of full travel at least once per 31 days was as follows (see Appendix C):

"CONCERN 1.

Commonwealth Edison's response in paragraph 3 does not contain the requirement of the Model Technical Specifications of paragraph 4.1.3.1.1b to cycle each valve at least one complete cycle of full travel at least once per 31 days.[*]

REQUEST 1.

Provide technical bases why the requested change is not applicable to Dresden Nuclear Power Station Units 2 and 3.[**]

Response:

The model Technical Specifications that were used as a reference to develop our submittal were attached to the July 7, 1980 letter from D. Eisenhut to all operating BWR licensees. The Franklin Research Center model Technical Specifications for Section 4.1.3.1.1 are not the same as the July 7, 1980 model Technical Specifications. In addition, the July 7, 1980, Model Technical Specifications were incorrect concerning the SDV vent/drain valve closure during individual CRD scram timing. The July 7, 1980, model is performed each refueling. A possible means of modifying our submittal would be to require verification of valve closure and subsequent re-opening during each scram, and take credit for that. In summary it is our contention that our proposal is meaningful and provides a true test of the system."

The Licensee agreed to revise the proposed specification changes to require cycling each valve at least one complete cycle of full travel at least once per quarter. .

EVALUATION

The revised page 3.3/4.3-3 of the Quad Cities Station Units 1 and 2 Technical Specifications with the agreed-upon revision complies with the requirement of paragraphs 4.1.3.1.1a and 4.1.3.1.1b of the NRC Staff's Model Technical Specifications which require verifying each valve to be open at least once per 31 days and cycling each valve at least one complete cycle of full travel at least once per quarter, respectively.

*On 10/22/81, the paragraph 4.1.3.1.1b was revised by the NRC: "Once per 31 days" was replaced by "once per 92 days."

**Appendix C is also applicable to Quad Cities Station Units 1 and 2.

4.2 LCO/SURVEILLANCE REQUIREMENTS FOR REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 3.3.1 and Table 3.3.1-1 require the functional unit of SDV water level-high to have at least 2 operable channels containing 2 limit switches per trip system, a total of 4 operable channels containing 4 limit switches per 2 trip systems for the reactor protection system which automatically initiates scram.

Paragraph 3.3.1 and Table 3.3.1-2 concern the response time of the reactor protection system for the functional unit of SDV water level-high which should be specified for each BWR (it is not specified in the table). Paragraph 4.3.1.1 and Table 4.3.1.1-1 require that each reactor protection system instrumentation channel containing a limit switch be shown to be operable for the functional unit of SDV water level-high by the Channel Functional Test monthly and Channel Calibration at each refueling outage. The applicable operational conditions for these requirements are startup, run, and refuel.

LICENSEE RESPONSE

Pages 3.1/4.1-8, 3.1/4.1-9, 3.1/4.1-10, and 3.1/4.1-11 of the existing Quad Cities Station Units 1 and 2 Technical Specifications address the NRC staff's Model Technical Specifications requirements of paragraph 3.3.1 and Table 3.3.1-1 by providing Table 3.1-1, "Reactor Protection System (Scram) Instrumentation Requirements Refuel Mode," Table 3.1-2, "Reactor Protection System (Scram) Instrumentation Requirements Startup/Hot Standby Mode," Table 3.1-3, "Reactor Protection System (Scram) Instrumentation Requirements Run Mode," and Table 3.1-4, notes for Tables 3.1-1, 3.1-2, and 3.1-3 with the following information for "Trip Function High-water level in scram discharge volume (4)":

"1. Minimum Number of Operable or Tripped Instrument Channels per Trip System (1): 2

2. Trip Level Setting: $\leq$ 50 gallons

3. Action (2): A

4. Action Required When Equipment Operability is Not Assured (1): A

NOTES:

1. There shall be two operable trip systems or one operable and one tripped system for each function

2. If the first column cannot be met for one of the trip systems, that trip system shall be tripped. If the first column cannot be met for both trip systems, the appropriate actions listed below shall be taken:

   A. Initiate insertion of operable rods and complete insertion of all operable rods within 4 hours.

4. Permissible to bypass, with control rod block for reactor protection system reset in refuel and shutdown positions of the reactor mode switch."

The requirements of paragraph 3.3.1 and Table 3.3.1-2 of the NRC staff's Model Technical Specifications are covered by page 3.3/4.3-10 of the Quad Cities Station Units 1 and 2 Technical Specifications which give the reactor protection system response time as follows:

"In the analytical treatment of the transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid deenergizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses and is also included in the allowable scram insertion times specified in Specification 3.3.C. The scram times for all control rods will be determined at the time of each refueling outage. A representative sample of control rods will be tested following a shutdown. Scram times of new drives are approximately 2.5 to 3 seconds; lower rates of change in scram times following initial plant operation at power are expected. The test schedule provides reasonable assurance of detection of slow drives before system deterioration beyond limits of Specification 3.3.C."

Pages 3.1/4.1-12, 3.1/4.1-13, and 3.1/4.1-14 of the present Quad Cities Station Units 1 and 2 Technical Specifications address the NRC staff's Model

Technical Specifications requirements of paragraph 4.3.1.1 and Table 4.3.1.1-1. Pages 3.1/4.1-12 and 3.1/4.1-13 contain Table 4.1-1, "Scram Instrumentation and Logic Systems Functional Tests Minimum Functional Test Frequencies for Safety Instrumentation, Logic Systems, and Control Circuits," which provides the following information for "Instrument Channel High water level in scram discharge volume":

"1. Group (3): A

2. Functional Test (7): Trip channel and alarm

3. Minimum Frequency (4): Every 3 months

NOTES:

3. A description of the three groups is included in the bases of this specification

   A. On-off sensors that provide a scram trip function

4. Functional tests are not required when the systems are not required to be operable or are tripped. If test are missed, they shall be performed prior to returning the systems to an operable status

7. A functional test of the logic of each channel is performed as indicated. This coupled with placing the mode switch in shutdown each refueling outage constitutes a logic system functional test of the scram system."

Page 3.1/4.1-14 contains Table 4.1-2, "Scram Instrument Calibration Minimum Calibration Frequencies for Reactor Protection Instrument Channels." The minimum calibration frequency for Instrument Channel High water level in scram discharge volume should be listed in this table. It is not. The Licensee's response to the RAI regarding the reactor protection system SDV Channel Functional Test and Channel Calibration is given below (see Appendix C).

"REQUEST 2.

The technical Specifications for Dresden 2 and 3[*] state that each reactor protection system scram discharge volume water level-high instrumentation channel containing a limit switch shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST once per 3 months, and there are no tech specs for CHANNEL CALIBRATION. Since the proposed frequency of the required surveillance for Dresden Nuclear Power Station Units 2 and 3 differs from the frequency requested by the Model Technical Specifications, provide technical bases for it.

RESPONSE

The proposed Technical Specifications regarding the SDV scram and rod block level switches are adequate. A monthly functional test of the SDV scram bypass would require the reactor mode switch to be placed in either SHUTDOWN or REFUEL for the test, and this is unreasonable.

REQUEST 3.

Provide technical bases for not calibrating the scram discharge volume water level-high instrumentation channel. Also provide technical basis for performing the scram trip bypassed instrumentation channel functional test once per refueling outage instead of once per month as requested in the Model Technical Specifications.

RESPONSE

Magnetrol level switches are not, and cannot, be calibrated. Therefore, calibration frequency in our submittal is designated as 'Not Applicable' for the scram discharge volume water level high channel."

The Licensee is installing a second instrument volume containing four additional limit switches, for a total of eight limit switches for the reactor protection system. In addition, the Licensee agreed to revise page 3.1/4.1-14 of the Quad Cities Station Units 1 and 2 Technical Specifications to incorporate in Table 4.1-2 the Calibration Test each refueling for "Instrument Channel-SDV Water Level High." The Calibration Test will consist of physical inspection and actuation of the level switches using water columns.

EVALUATION

Pages 3.1/4.1-8 through 3.1/4.1-11 of the existing Quad Cities Station Units 1 and 2 meet the NRC staff's Model Technical Specifications requirements

*Appendix C is also applicable to Quad Cities Station Units 1 and 2.

of paragraph 3.3.1 and Table 3.3.1-1 in regard to the minimum number of operable instrument channels per trip system and number of trip systems, and are acceptable. The Quad Cities Station Units 1 and 2 reactor protection system SDV water level-high instrumentation consists of 2 operable channels containing 2 limit switches per trip system, for a total of 4 operable channels containing 4 limit switches per 2 trip systems, making 1-out-of-2-taken-twice logic. The specified trip level setting of $\leq$ 50 gallons for scram initiation and operating modes of refuel, startup/hot standby, and run are also acceptable.

The reactor protection system response time of 290 milliseconds specified on original page 3.3/4.3-10 of the Quad Cities Station Units 1 and 2 Technical Specifications is acceptable and addresses the requirements of paragraph 3.3.1 and Table 3.3.1-2.

The provision of the present Quad Cities Station Units 1 and 2 Technical Specifications given on page 3.1/4.1-12, Table 4.1-1, for a reactor protection system SDV water level-high Channel Functional Test once per 3 months does not meet the NRC staff's Model Technical Specifications requirement of paragraph 4.3.1.1 and Table 4.3.1.1-1 for the Channel Functional Test to be performed monthly. However, the Licensee is installing a second instrument volume containing four additional limit switches, for a total of eight limit switches for the reactor protection system. This increases significantly the reliability of the system and provides technical bases for acceptance of the proposed surveillance requirements to perform Channel Functional Test quarterly.

The Licensee agreed to revise page 3.1/4.1-14 of the Quad Cities Station Units 1 and 2 Technical Specifications to incorporate in Table 4.1-2 the Calibration Test "each refueling" for "Instrument Channel - SDV water level high." The Calibration Test will consist of physical inspection and actuation of the level switches using water columns. This meets the NRC staff's Model Technical Specifications requirements of paragraph 4.3.1.1 and Table 4.3.1.1-1 and is acceptable.

4.3 LCO/SURVEILLANCE REQUIREMENTS FOR CONTROL ROD WITHDRAWAL BLOCK SDV LIMIT SWITCHES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 3.3.6 and Table 3.3.6-1 require the control rod withdrawal block instrumentation to have at least 2 operable channels containing 2 limit switches for SDV water level-high and 1 operable channel containing 1 limit switch for SDV trip bypassed. Paragraph 3.3.6 also requires specifying the trip setpoint for control rod withdrawal block instrumentation monitoring SDV water level-high as indicated in Table 3.3.6-2.

Paragraph 4.3.6 and Table 4.3.6-1 require each control rod withdrawal block instrumentation channel containing a limit switch to be shown to be operable by the Channel Functional Test once per 3 months for SDV water level-high, by the Channel Functional Test once per month for SDV scram trip bypassed, and by Channel Calibration at each refueling outage for SDV water level-high.

LICENSEE RESPONSE

The Licensee proposed to revise pages 3.2/4.2-14 and 3.2/4.2-16 of the Quad Cities Stations Units 1 and 2 Technical Specifications. On page 3.2/4.2-14, Table 3.2-3, "Instrumentation that Initiates Rod Block," provides the following information:

"Table 3.2-3

Minimum Number of Operable or Tripped Instrument Channels per Trip System (1)	Instrument	Trip Level Setting
1	High water level in scram discharge volume (SDV)	≤ 25 gallons
1	SDV high water level scram trip bypassed	NA*

NOTE: 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 system shall be tripped."

On page 3.2/4.2-16, Table 4.2-1, "Minimum Test and Calibration Frequency for Core and Containment Cooling Systems Instrumentation, Rod Blocks, and Isolations," line 12 was added as shown in the following table:

*This line was added in the revised edition.

"Table 4.2-1

Instrument Channel	Instrument Functional Test (2)	Calibration (2)	Instrument Check (2)
Rod Blocks			
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

NOTES: 2. Functional test, calibrations, and instrument checks are not required when those instruments are not required to be operable or are tripped."

The Licensee's response to the RAI, Requests 2 and 3, given in Section 4.2 of this report, is also applicable to this section.

In addition, the Licensee agreed to:

1. delete "Instrument Channel-SDV high water level scram trip bypassed" from Table 3.2-3 (revised page 3.2/4.2-14) and Table 4.2-1 (revised page 3.2/4.2-16)

2. revise the first sentence of Note 1, Table 3.2-3 on page 3.2/4.2-14, to state the following or its equivalent:

   "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 and 'Instrument-Rod Block, SDV high water level'."

3. incorporate into Table 4.2-1 (revised page 3.2/4.2-16) the Calibration Test "each refueling" instead of "Not applicable" for "Instrument-Rod Block, High water level in scram discharge volume."

EVALUATION

The existing Quad Cities Station Units 1 and 2 scram discharge system has six level switches on the scram discharge volume (see FSAR page 3.5-5 and Section 10.6) set at three different water levels to guard against operation of the reactor without sufficient free volume present in the scram discharge headers to receive the scram discharge water in the event of a scram. At the first (lowest) level with a setpoint of 3 gallons (see FSAR Table 7.7.1, "Typical Protection Systems Setpoints"), one level switch initiates an alarm for operator action. At the second level with a setpoint of ≤ 25 gallons (see the present Quad Cities Technical Specifications, page 3.2/4.2-14, Table 3.2-3), one level switch* initiates a rod withdrawal block to prevent further withdrawal of any control rod. At the third (highest) level with a setpoint of 50 gallons (see FSAR, Table 7.7.1), the four level switches (two for each reactor protection system trip system) initiate a scram to shut down the reactor while sufficient free volume is available to receive the scram discharge water. Reference 9, page 50, defines Design Criterion 9 ("Instrumentation shall be provided to aid the operator in the detection of water accumulation in the instrumented volume(s) prior to scram initiation"), gives the technical basis for "Long-Term Evaluation of Scram Discharge System," and defines acceptable compliance ("The present alarm and rod block instrumentation meets this criterion given adequate hydraulic coupling with the SDV headers"). Thus, if the Quad Cities Station Units 1 and 2 scram discharge system is modified (long term) so that the hydraulic coupling between scram discharge headers and instrumented volume is adequate and acceptable, then the present alarm and rod block instrumentation consisting of one trip system with one instrument channel containing one limit switch is also acceptable.

When the reactor of Quad Cities Station Units 1 and 2 is in operational conditions of startup and run, "Scram Discharge Volume Scram Trip" cannot be bypassed, and operational condition "refuel with more than one control rod withdrawn" is not applicable (see FSAR, page 3.5-5: "Interlocks are provided which prevent the inadvertent withdrawal of more than one control rod with the mode switch in the refuel position"). Thus, the NRC staff's Model Technical Specifications requirements of paragraph 3.3.6, Table 3.3.6-1, paragraph 4.3.6, and Table 4.3.6-1 for "Trip Function 5. b. Scram Discharge Volume Scram Trip

*The existing SDV system in regard to control rod withdrawal block SDV limit switches does not comply with the present Quad Cities Technical Specifications which require two trip systems with two level switches (see page 3.2/4.2-14, Table 3.2-3, and Note 1).

Bypassed" are not applicable to Quad Cities Station Units 1 and 2 for the specified operational conditions. Therefore, the Licensee agreed to delete "Instrument Channel-SDV high water level scram trip bypassed" from the proposed revision of page 3.2/4.2-14, Table 3.2-3, and page 3.2/4.2-16, Table 4.2-1.

Since the existing system has only one trip system with one instrument channel containing one control rod withdrawal block SDV limit switch and is acceptable, to reflect this, the Licensee agreed to revise the first sentence of Note 1 in Table 3.2-3 on page 3.2/4.2-14 as follows or its equivalent: "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 and 'Instrument-Rod Block SDV high water level'."

The specified Trip Level Setting of ≤ 25 gallons in Table 3.2.3, page 3.2/4.2-14, for control rod withdrawal block instrumentation, monitoring SDV water level-high, is acceptable. It meets the NRC staff's Model Technical Specifications requirements of paragraph 3.3.6 and Table 3.3.6-2.

Since the Licensee agreed to incorporate into Table 4.2-1, on revised page 3.2/4.2-16, the Calibration Test "each refueling" instead of "Not applicable" for "Instrument-Rod Block, high water level in scram discharge volume," the revised Table 4.2-1 will comply with the NRC staff's Model Technical Specifications requirements of paragraph 4.3.6 and Table 4.3.6-1 for the Channel Functional Test once per 3 months and Channel Calibration each refueling for control rod withdrawal block SDV water level-high. The Channel Calibration will consist of physical inspection and actuation of the level switch using water column.

5. CONCLUSIONS

Table 5-1 summarizes the results of the final review and evaluation of the Quad Cities Station Units 1 and 2 Phase 1 proposed Technical Specifications changes for SDV long-term modification in regard to surveillance requirements for SDV vent and drain valves and LCO/surveillance requirements for reactor protection system and control rod block SDV limit switches. The following conclusions were made:

- The revised page 3.3/4.3-3, with the Licensee's agreement to incorporate a revision into the proposed specifications changes that requires cycling each valve at least one complete cycle of full travel at least quarterly, complies with the NRC staff's Model Technical Specifications requirements of paragraphs 4.1.3.1.1a and 4.1.3.1.1b.

- The original pages 3.1/4.1-12 and 3.1/4.1-13, Table 4.1-1, of the Quad Cities Station Units 1 and 2 Technical Specifications, which provide for the reactor protection system SDV limit switches water level-high Channel Functional Test to be performed once per 3 months, do not meet the surveillance requirement (paragraph 4.3.1.1, Table 4.3.1.1-1, of the NRC staff's Model Technical Specifications) for the test to be performed monthly. However, the Licensee is installing a second instrument volume containing four additional limit switches, for a total of eight limit switches, for the reactor protection system. This increases significantly the reliability of the system and provides technical bases for acceptance of the proposed surveillance requirements to perform the Channel Functional Test quarterly.

- To meet the NRC staff's Model Technical Specifications requirements of paragraph 4.3.1.1 and Table 4.3.1.1-1, the Licensee agreed to revise page 3.1/4.1-14 of the present Quad Cities Station Units 1 and 2 Technical Specifications to incorporate into Table 4.1-2 the Calibration Test "each refueling" for "Instrument Channel - SDV Water Level High." The Calibration Test will consist of physical inspection and actuation of the level switches using water columns.

- The NRC staff's Model Technical Specifications surveillance requirements in paragraph 3.3.6, Table 3.3.6-1, paragraph 4.3.6, and Table 4.3.6-1 for control rod block SDV scram trip bypassed are not applicable to the operational conditions of startup, run, and refuel with more than one control rod withdrawn. Therefore, the Licensee agreed to delete "Instrument Channel- SDV high water level scram trip bypassed" from the proposed revision of page 3.2/4.2-14, Table 3.2-3, and page 3.2/4.2-16, Table 4.2-1.

- o The existing SDV system has only one trip system with one instrument channel containing one control rod withdrawal block SDV limit switch and meets NRC criteria. To reflect this, the Licensee agreed to revise the first sentence of Note 1, in Table 3.2-3 on original page 3.2/4.2-14.

- o To meet the NRC staff's Model Technical Specifications requirements of paragraph 4.3.6 and Table 4.3.6-1, the Licensee agreed to incorporate into Table 4.2-1 on revised page 3.2/4.2-16 the Calibration Test "each refueling" instead of "Not applicable" for instrument Channel-Rod Blocks, high water level in scram discharge volume." Channel Calibration with the Magnetrol level switch will consist of physical inspection and actuation of the switch using a water column.

- o The remaining surveillance requirements are met by revised pages 3.2/4.2-14, 3.2/4.2-16, 3.3/4.3-3, 3.3/4.3-9, and original, unrevised pages 3.1/4.1-12, 3.1/4.1-13, 3.1/4.1-14, and 3.3/4.3-10 of the Quad Cities Station Units 1 and 2 Technical Specifications.

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Table 5-1 Evaluation of Phase 1 Proposed Technical Specifications Changes for Scram Discharge Volume Long-Term Modifications Quad Cities Station Units 1 and 2

Surveillance Requirements	Technical Specifications: NRC Staff Model (Paragraph)	Technical Specifications: Proposed by Licensee	Evaluation
SDV DRAIN AND VENT VALVES			
Verify each valve open	Once per 31 days (4.1.3.1.1a)	Once per 31 days (pp. 3.3/4.3-3 and 3.3/4.3-9 revised)	Acceptable
Cycle each valve one complete cycle	Once per 92 days (4.1.3.1.1b)	Once per quarter (p. 3.3/4.3-3, second revision)	Acceptable
REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES			
Minimum operable channels per trip system	2 (3.3.1, Table 3.3.1-1)	2 (pp. 3.1/4.1-8 to 3.1/4.1-11)	Acceptable
SDV water level-high response time	NA (3.3.1, Table 3.3.1-2)	0.290 sec max. (p. 3.3/4.3-10)	Acceptable
SDV water level-high			
Channel functional test	Monthly (4.3.1.1, Table 4.3.1.1-1)	Every 3 months (pp. 3.1/4.1-12 and 3.1/4.1-13)	Acceptable (see p. 20 of this TER)
Channel calibration	Each refueling (4.3.1.1, Table 4.3.1.1-1)	Each refueling (p. 3.1/4.1-14, Table 4.1-2, to be revised)	Acceptable

Table 5-1 (Cont.)

Surveillance Requirements	Technical Specifications: NRC Staff Model (Paragraph)	Technical Specifications: Proposed by Licensee	Evaluation
CONTROL ROD BLOCK SDV LIMIT SWITCHES			
Minimum operable channels per trip function			
SDV water level-high	2 (3.3.6, Table 3.3.6-1)	1 (p. 3.2/4.2-14, Table 3.2-3, revised)	Acceptable*
SDV scram trip bypassed	1 (3.3.6, Table 3.3.6-1)	Not applicable (p. 3.2/4.2-14, Table 3.2-3, second revision)	Acceptable*
SDV water level-high			
Trip setpoint	NA (3.3.6, Table 3.3.6-2)	≤ 25 gallons (p. 3.2/4.2-14, Table 3.2-3, revised)	Acceptable
Channel functional test	Quarterly (4.3.6, Table 4.3.6-1)	Once per 3 months (p. 3.2/4.2-16, Table 4.2-1, revised)	Acceptable
Channel calibration	Each refueling (4.3.6, Table 4.3.6-1)	Each refueling (p. 3.2/4.2-16, Table 4.2-1, second revision)	Acceptable
SDV scram trip bypassed			
Channel functional test	Monthly (4.3.6, Table 4.3.6-1)	Not applicable (p. 3.2/4.2-16, Table 4.2-1, second revision)	Acceptable*

* See Reference 9, p. 50, and pp. 21 to 25 of this TER.

6. REFERENCES

1. "Degradation of BWR Scram Discharge Volume Capability"
   NRC, Office of Inspection and Enforcement, June 12, 1980
   IE Bulletin 80-14

2. D. G. Eisenhut (NRR)
   Letter "To All Operating Boiling Water Reactors (BWRs)" with enclosure, "Model Technical Specifications"
   July 7, 1980

3. "Failure of 76 of 185 Control Rods to Fully Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 3, 1980
   IE Bulletin 80-17

4. Supplement 1, "Failure of 76 of 185 Control Rods to Fully Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 18, 1980
   IE Bulletin 80-17

5. Supplement 2, "Failures Revealed by Testing Subsequent to Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 22, 1980
   IE Bulletin 80-17

6. Supplement 3, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, August 22, 1980
   IE Bulletin 80-17

7. Supplement 4, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, December 18, 1980
   IE Bulletin 80-17

8. Supplement 5, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, February 13, 1981
   IE Bulletin 80-17

9. P. S. Check (NRR)
   Memorandum with enclosure, "Generic Safety Evaluation Report BWR Scram Discharge System"
   December 1, 1980

10. P. S. Check (NRR)
    Memorandum with enclosure, "Staff Report and Evaluation of Supplement 4 to IE Bulletin 80-17"
    June 10, 1981

APPENDIX A

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

* Note: Applicable changes are marked by vertical lines in the margins.

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. If the inoperable control rod(s) is inserted, within one hour disarm the associated directional control valves either:
   a) Electrically, or
   b) Hydraulically by closing the drive water and exhaust water isolation valves.
3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. Verifying each valve to be open* at least once per 31 days and
b. Cycling each valve through at least one complete cycle of full travel at least once per 92 days.

4.1.3.1.2 When above the preset power level of the RWM and RSCS, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 7 days, and
b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.

*These valves may be closed intermittently for testing under administrative controls.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position (6), based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed (7.0) seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding (7.0) seconds:

a. Declare the control rod(s) with the slow insertion time inoperable, and

b. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of (7.0) seconds, or

c. Be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,

b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

c. For 10% of the control rods, on a rotating basis, at least once per 120 days of operation.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place at least one inoperable channel in the tripped condition within one hour.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one inoperable channel in at least one trip system* in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip function shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function.

* If both channels are inoperable in one trip system, select at least one inoperable channel in that trip system to place in the tripped condition, except when this would cause the Trip Function to occur.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
8. Scram Discharge Volume Water Level - High	1, 2, 5[h]	2	4
9. Turbine Stop Valve - Closure	1[i]	4[j]	7
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	1[i]	2[j]	7
11. Reactor Mode Switch in Shutdown Position	1, 2, 3, 4, 5	1	8
12. Manual Scram	1, 2, 3, 4, 5	1	9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - In OPERATIONAL CONDITION 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 2 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Be in at least STARTUP within 2 hours.

ACTION 4 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 5 - Be in at least HOT SHUTDOWN within 6 hours.

ACTION 6 - Be in STARTUP with the main steam line isolation valves closed within 2 hours or in at least HOT SHUTDOWN within 6 hours.

ACTION 7 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to < (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER, within 2 hours..

ACTION 8 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, verify all insertable control rods to be fully inserted within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 9 In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, lock the reactor mode switch in the Shutdown position within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

*Except movement of IRM, SRM or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than (11) LPRM inputs to an APRM channel.

(d) These functions are not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(g) Also actuates the standby gas treatment system.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(i) These functions are automatically bypassed when turbine first stage pressure is ≤ (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER.

(j) Also actuates the EOC-RPT system.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - Upscale	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, (15)%	NA
b. Flow Biased Simulated Thermal Power - Upscale	$\leq$ (0.09)**
c. Fixed Neutron Flux - Upscale, (118)%	$\leq$ (0.09)
d. Inoperative	NA
e. LPRM	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ (0.55)
4. Reactor Vessel Water Level - Low, Level 3	$\leq$ (1.05)
5. Main Steam Line Isolation Valve - Closure	$\leq$ (0.06)
6. Main Steam Line Radiation - High	NA
7. Primary Containment Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Stop Valve - Closure	$\leq$ (0.06)
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\leq$ (0.08)#
11. Reactor Mode Switch in Shutdown Position	NA
12. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel. (This provision is not applicable to Construction Permits docketed after January 1, 1978. See Regulatory Guide 1.18, November 1977.)
**Not including simulated thermal power time constant.
#Measured from start of turbine control valve fast closure.

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TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	NA	M	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	M	R	1
10. Turbine Control Valve Fast Closure Trip Oil Pressure - Low	NA	M	Q	1
11. Reactor Mode Switch in Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) Within 24 hours prior to startup, if not performed within the previous 7 days.
(c) The IRM and SRM channels shall be determined to overlap for at least ( ) decades during each startup and the IRM and APRM channels shall be determined to overlap for at least ( ) decades during each controlled shutdown, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference greater than 2%. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM readout to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

INSTRUMENTATION

3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod withdrawal block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistant with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

TABLE 3.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD BLOCK MONITOR[(a)]			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. APRM			
a. Flow Biased Simulated Thermal Power - Upscale	4	1	61
b. Inoperative	4	1, 2, 5	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5	61
3. SOURCE RANGE MONITORS			
a. Detector not full in(b)	3	2	61
	2	5	61
b. Upscale[(c)]	3	2	61
	2	5	61
c. Inoperative[(c)]	3	2	61
	2	5	61
d. Downscale[(d)]	3	2	61
	2	5	61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in (e)	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative[(e)]	6	2, 5	61
d. Downscale	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5**	62
b. Scram Trip Bypassed	1	(1, 2, 5**)	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. (Comparator) (Downscale)	2	1	62

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TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

NOTES

* With THERMAL POWER $\geq$ (20)% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

a. The RBM shall be automatically bypassed when a peripheral control rod is selected.

b. This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range (2) or higher.

c. This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

d. This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

e. This function shall be automatically bypassed when the IRM channels are on range 1.

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TABLE 3.3.6-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	≤ 0.66 W + (40)%	≤ 0.66 W + (43)%
b. Inoperative	NA	NA
c. Downscale	≥ (5)% of RATED THERMAL POWER	≥ (3)% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Simulated Thermal Power - Upscale	≤ 0.66 W + (42)%*	≤ 0.66 W + (45)%*
b. Inoperative	NA	NA
c. Downscale	≥ (5)% of RATED THERMAL POWER	≥ (3)% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	≤ (12)% of RATED THERMAL POWER	≤ (14)% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ (2 x 10^5) cps	≤ (5 x 10^5) cps
c. Inoperative	NA	NA
d. Downscale	≥ (3) cps	≥ (2) cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ (108/125) of full scale	≤ (110/125) of full scale
c. Inoperative	NA	NA
d. Downscale	≥ (5/125) of full scale	≥ (3/125) of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level High	To be specified	NA
b. Scram Trip Bypassed	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ (___/___) of full scale	≤ (___/___) of full scale
b. Inoperative	NA	NA
c. (Comparator) (Downscale)	≤ (10)% flow deviation	≤ (__)% flow deviation

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

TABLE 4.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U(b),M	Q	1*
b. Inoperative	NA	S/U(b),M	NA	1*
c. Downscale	NA	S/U(b),M	Q	1*
2. APRM				
a. Flow Biased Simulated Thermal Power - Upscale	NA	S/U(b),M	Q	1
b. Inoperative	NA	S/U(b),M	NA	1, 2, 5
c. Downscale	NA	S/U(b),M	Q	1
d. Neutron Flux - Upscale, Startup	NA	S/U(b),M	Q	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U(b),W(c)	NA	2, 5
b. Upscale	NA	S/U(b),W(c)	Q	2, 5
c. Inoperative	NA	S/U(b),W(c)	NA	2, 5
d. Downscale	NA	S/U(b),W(c)	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U(b),W(c)	NA	2, 5
b. Upscale	NA	S/U(b),W(c)	Q	2, 5
c. Inoperative	NA	S/U(b),W(c)	NA	2, 5
d. Downscale	NA	S/U(b),W(c)	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Trip Bypassed	NA	M	NA	(1, 2, 5**)
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U(b),M	Q	1
b. Inoperative	NA	S/U(b),M	NA	1
c. (Comparator) (Downscale)	NA	S/U(b),M	Q	1

TABLE 4.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 24 hours prior to startup, if not performed within the previous 7 days.

c. When making an unscheduled change from OPERATIONAL CONDITION 1 to OPERATIONAL CONDITION 2, perform the required surveillance within 12 hours after entering OPERATIONAL CONDITION 2.

* With THERMAL POWER ≥ (20)% of RATED THERMAL POWER.

** With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

APPENDIX B

COMMONWEALTH EDISON LETTER OF OCTOBER 14, 1980

AND

SUBMITTAL WITH PROPOSED TECHNICAL SPECIFICATIONS CHANGES

FOR

QUAD CITIES STATION UNITS 1 AND 2

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8010240203 DOC.DATE: 80/10/14 NOTARIZED: YES DOCKET #
FACIL:50-237 Dresden Nuclear Power Station, Unit 2, Commonwealth E 05000237
50-249 Dresden Nuclear Power Station, Unit 3, Commonwealth E 05000249
50-254 Quad-Cities Station, Unit 1, Commonwealth Edison Co. 05000254
50-265 Quad-Cities Station, Unit 2, Commonwealth Edison Co. 05000265
AUTH.NAME AUTHOR AFFILIATION
JANECEK,R.F. Commonwealth Edison Co.
RECIP.NAME RECIPIENT AFFILIATION
Office of Nuclear Reactor Regulation, Director

SUBJECT: Forwards proposed Tech Spec changes,per NRC 800707 request. Changes provide addl operational & surveillance requirements.

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October 14, 1980

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Dresden Station Units 2 and 3
Quad Cities Station Units 1 and 2
Proposed Amendment to Appendix A,
Technical Specifications, to
Operating Licenses DPR-19, 25, 29, and 30
NRC Docket Nos. 50-237/249 and 50-254/265

Reference (a): D. G. Eisenhut letter to All Operating
Boiling Water Reactors (BWR's) dated
July 7, 1980

Dear Sir:

In accordance with the request in Reference (a), and pursuant to 10 CFR 50.59, Commonwealth Edison proposes to amend Appendix A, Technical Specifications, to Operating Licenses DPR-19, DPR-25, DPR-29, and DPR-30 for Dresden Units 2, 3 and Quad Cities Units 1, 2, respectively. The proposed amendments would add surveillance requirements for the scram discharge volume (SDV) vent and drain valves and LCO/surveillance requirements for RPS and control rod block SDV limit switches.

The proposed changes provide additional operational and surveillance requirements and thereby strengthen the provisions for assuring continued operability of the control rod drive system during reactor operation. As such, the proposed changes do not present an unreviewed safety concern nor do they present any additional hazard to the health and safety of the public.

The proposed changes were prepared in accordance with the guidance provided in Reference (a) and have received On-Site and Off-Site review and approval. The changes are included in Attachments 1, 2, 3, and 4 for Dresden 2, Dresden 3, Quad Cities 1, and Quad Cities 2, respectively.

Pursuant to 10 CFR 170, Commonwealth Edison has reviewed the proposed changes and determined them to be two (2) Class III and two (2) Class I Amendments. As such, a fee remittance in the amount of $8,800.00 has been provided.

8010240203 P

Please address any questions you may have concerning this matter to this office.

Three (3) signed originals and fifty-nine (59) copies of this transmittal are provided for your use.

Very truly yours,

Robert F. Janecek

Robert F. Janecek
Nuclear Licensing Administrator
Boiling Water Reactors

SUBSCRIBED and SWORN to
before me this 14th,
day of [illegible], 1980

[illegible]
Notary Public

cc: RIII Inspector, Dresden
RIII Inspector, Quad Cities

7283A

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

QUAD-CITIES
DPR-29

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)	≤0.58W + 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)	≤0.65W + 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	≤10⁵ counts/sec
2(5)	SRM downscale(9)	≥10² 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.

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QUAD-CITIES

DPR-29

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

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QUAD CITIES
DPR-29

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.

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. 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.

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. 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

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.

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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 or 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

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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]	≤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), 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. The 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.

3.2/4.2-14

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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

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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.

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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 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.

APPENDIX C

COMMONWEALTH EDISON LETTER OF OCTOBER 22, 1981

WITH

ANSWER TO RFI

FOR

QUAD CITIES STATION UNITS 1 AND 2

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

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50-249 Dresden Nuclear Power Station, Unit 3, Commonwealth E 05000249
50-254 Quad-Cities Station, Unit 1, Commonwealth Edison Co. 05000254
50-265 Quad-Cities Station, Unit 2, Commonwealth Edison Co. 05000265
AUTH.NAME AUTHOR AFFILIATION
RAUSCH,T.J. Commonwealth Edison Co.
RECIP.NAME RECIPIENT AFFILIATION
IPPOLITO,T.A. Operating Reactors Branch 2

SUBJECT: Forwards response to NRC 811902 request for addl info re proposed Tech Spec changes for scram discharge vol vent & drain valves.

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October 22, 1981

RECEIVED
OCT 23 1981
U.S. NUCLEAR REGULATORY COMMISSION

Mr. T. A. Ippolito, Chief
Operating Reactors - Branch 2
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Dresden Station Units 2 and 3
Quad Cities Station Units 1 and 2
Response to Request for Information
Concerning SDV Vent and Drain
Valve Technical Specifications
NRC Docket Nos. 50-237/249 and
50-254/265

References (a): T. A. Ippolito letter to L. O. DelGeorge dated September 2, 1981

(b): R. F. Janecek letter to Director of NRR dated October 14, 1980

Dear Mr. Ippolito:

Commonwealth Edison has received your Reference (a) request for additional information concerning our Reference (b) proposed Technical Specification changes pertaining to Scram Discharge Volume (SDV) vent and Drain Valves. Although you only requested information for Dresden Units 2 and 3, our response includes Quad Cities Units 1 and 2 as well, since the questions are directly applicable.

Your requests and our responses are provided in the attachment to this letter.

Please address any further questons you may have in this regard to this office.

One (1) signed original and fifty-nine (59) copies of this transmittal are provided for your use.

Very truly yours,

Thomas J Rausch

Thomas J. Rausch
Nuclear Licensing Administrator
Boiling Water Reactors

Attachment

cc: RIII Inspector, Dresden
RIII Inspector, Quad Cities

2730N

A001
s/11

8110290110 811022
PDR ADOCK 05000237
P PDR

Attachment
Commonwealth Edison Company
Dresden Units 2 and 3
Quad Cities Units 1 and 2

Response to Request for Information Concering
SDV Vent and Drain Valve Technical Specifications

CONCERN 1.

Commonwealth Edison's response in paragraph 3 does not contain the requirement of the Model Technical Specifications of paragraph 4.1.3.1.1b to cycle each valve at least one complete cycle of full travel at least once per 31 days.

REQUEST 1.

Provide technical bases why the requested change is not applicable to Dresden Nuclear Power Station Units 2 and 3.

Response:

The model Technical Specifications that were used as a reference to develop our submittal were attached to the July 7, 1980 letter from D. Eisenhut to all operating BWR licensees. The Franklin Research Center model Tennical Specifications for Section 4.1.3.1.1 are not the same as the July 7, 1980 model Technical Specifications. In addition, the July 7, 1980, model Technical Specifications were incorrect concerning the SDV vent/drain valve closure during individual CRD scram timing. The July 7, 1980, model Technical Specifications were modified to ensure a meaningful test is performed each refueling. A possible means of modifying our submittal would be to require verification of valve closure and subsequent re-opening during each scram, and take credit for that. In summary it is our contention that our proposal is meaningful and provides a true test of the system.

REQUEST 2.

The Technical Specifications for Dresden 2 and 3 state that each reactor protection system scram discharge volume water level-high instrumentation channel containing a limit switch shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST once per 3 months, and there are not tech specs for CHANNEL CALIBRATION. Since the proposed frequency of the required surveillance for Dresden Nuclear Power Station Units 2 and 3 differs from the frequency requested by the Model Technical Specifications, provide technical bases for it.

RESPONSE

The proposed Technical Specifications regarding the SDV scram and rod block level switches are adequate. A monthly functional test of the SDV scram bypass would require the reactor mode switch to be placed in either SHUTDOWN or REFUEL for the test, and this unreasonable.

REQUEST 3.

Provide technical basis for not calibrating the scram discharge volume water level-high instrumentation channel. Also provide technical basis for performing the scram trip bypassed instrumentation channel functional test once per refueling outage instead of once per month as requested in the Model Technical Specifications.

RESPONSE

Magnetrol level switches are not, and cannot, be calibrated. Therefore, calibration frequency in our submittal is designated as "Not Applicable" for the scram discharge volume water level high channel.

2730N

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-254 AND 50-265

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO

FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 84 and 77 to Facility Operating License Nos. DPR-29 and DPR-30, issued to Commonwealth Edison Company and Iowa-Illinois Gas and Electric Company, which revised the Technical Specifications for operation of the Quad-Cities Nuclear Power Station, Units 1 and 2, located in Rock Island County, Illinois. The amendments are effective as of the date of issuance.

The amendments expand the Technical Specifications for the scram discharge volume (SDV) to include surveillance requirements for SDV vent and surviellance requirements for the reactor protection system (RPS) and control rod block SDV limit switches.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of the amendments was not required since they do not involve a significant hazards consideration.

The Commission has determined that the issuance of the amendments will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendments.

For further details with respect to this action, see (1) the application for amendments dated October 14,1980, as supplemented October 22, 1981, (2) Amendment Nos. 84 and 77 to License Nos. DPR-29 and DPR-30 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room 1717 H Street N. W. Washington, D. C., and at the Moline Public Library, 504 - 17th Street, Moline, Illinois. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C., 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 23rd day of December 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing