March 28, 2001

Mr. Oliver D. Kingsley, President
Exelon Nuclear
Exelon Generation Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS (TAC NOS. MA7793 AND MA7794)

Dear Mr. Kingsley:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 198 to Facility Operating License No. DPR-29 and Amendment No. 194 to Facility Operating License No. DPR-30 for the Quad Cities Nuclear Power Station, Units 1 and 2, respectively. The amendments are in response to your application, dated December 27, 1999. The application was submitted by Commonwealth Edison (ComEd), which has since merged to form Exelon Generation Company, LLC (EGC, the licensee). By letter dated February 7, 2001, EGC assumed responsibility for all pending Commission actions that were requested by ComEd.

The amendments revise Technical Specifications (TS) to increase allowable out-of-service times (AOTs) and surveillance test intervals (STIs) for selected instrumentation. The amendments implement AOT/STI changes based on Topical Reports by General Electric Company and the Boiling Water Reactor Owners' Group which the staff has previously reviewed and approved.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

Lawrence W. Rossbach, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-254 and 50-265

Enclosures:
1. Amendment No. 198 to DPR-29
2. Amendment No. 194 to DPR-30
3. Safety Evaluation

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NRR-058

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 28, 2001

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Exelon Nuclear
Exelon Generation Company, LLC
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cc w/encls: See next page

cc: w/enclosures

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 198
License No. DPR-29

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee, formerly Commonwealth Edison Company) dated December 27, 1999, 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 Operating 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. 198, 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 its issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 28, 2001

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 194
License No. DPR-30

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

    A. The application for amendment by Exelon Generation Company, LLC (the licensee, formerly Commonwealth Edison Company) dated December 27, 1999, 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 Operating 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. 194 , 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 its issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 28, 2001

ATTACHMENT TO LICENSE AMENDMENT NOS. 198 AND 194

FACILITY OPERATING LICENSE NOS. DPR-29 AND DPR-30

DOCKET NOS. 50-254 AND 50-265

Revise the Appendix "A" Technical Specifications by replacing the pages identified below with the attached pages. The revised pages are identified by amendment number and contain a vertical line indicating the area of change.

REMOVE	INSERT
1-8	1-8
3/4.1-1	3/4.1-1
3/4.1-6	3/4.1-6
3/4.1-7	3/4.1-7
3/4.1-8	3/4.1-8
3/4.1-9	3/4.1-9
3/4.1-10	3/4.1-10
B 3/4.1-1	B 3/4.1-1
3/4.2-1	3/4.2-1
3/4.2-2	3/4.2-2
3/4.2-7	3/4.2-7
3/4.2-8	3/4.2-8
3/4.2-9	3/4.2-9
3/4.2-10	3/4.2-10
3/4.2-11	3/4.2-11
3/4.2-12	3/4.2-12
3/4.2-14	3/4.2-14
3/4.2-15	3/4.2-15
3/4.2-16	3/4.2-16
-	3/4.2-16a
3/4.2-17	3/4.2-17
3/4.2-18	3/4.2-18
3/4.2-19	3/4.2-19
3/4.2-20	3/4.2-20
3/4.2-23	3/4.2-23
3/4.2-24	3/4.2-24
3/4.2-26	3/4.2-26
3/4.2-27	3/4.2-27
3/4.2-28	3/4.2-28
3/4.2-33	3/4.2-33
3/4.2-34	3/4.2-34
3/4.2-35	3/4.2-35
3/4.2-49	3/4.2-49
3/4.2-50	3/4.2-50
B 3/4.2-1	B 3/4.2-1
B 3/4.2-2	B 3/4.2-2
B 3/4.2-3	B 3/4.2-3
B 3/4.2-4	B 3/4.2-4
B 3/4.2-5	B 3/4.2-5
–	B 3/4.2-6
–	B 3/4.2-7

TABLE 1-1

SURVEILLANCE FREQUENCY NOTATION

	NOTATION	FREQUENCY
1. Shift	S	At least once per 12 hours
2. Day	D	At least once per 24 hours
3. Week	W	At least once per 7 days
4. Month	M	At least once per 31 days
5. Bimonthly	BIM	At least once per 60 days
6. Quarter	Q	At least once per 92 days
7. Semiannual	SA	At least once per 184 days
8. Annual	A	At least once per 366 days
9. Sesquiannual	E	At least once per 18 months (550 days)
10. Startup	S/U	Prior to each reactor startup
11. Not Applicable	NA	Not applicable

3.1 - LIMITING CONDITIONS FOR OPERATION

A. Reactor Protection System (RPS)

The reactor protection system (RPS) instrumentation CHANNEL(s) shown in Table 3.1.A-1 shall be OPERABLE.

APPLICABILITY:

As shown in Table 3.1.A-1.

ACTION:

1. With one CHANNEL required by Table 3.1.A-1 inoperable for Functional Units 1 through 12 in one or more Functional Units, place the inoperable CHANNEL and/or that TRIP SYSTEM in the tripped condition[a] within 12 hours.

2. With two or more CHANNELS required by Table 3.1.A-1 inoperable for Functional Units 1 through 12 in one or more Functional Units:
   a. Within one hour, verify sufficient CHANNELS remain OPERABLE or tripped[a] to maintain trip capability in the Functional Unit, and
   b. Within 6 hours, place the inoperable CHANNEL(s) in one TRIP SYSTEM and/or that TRIP SYSTEM[b] in the tripped condition[a], and
   c. Within 12 hours, restore the inoperable CHANNELS in the other TRIP SYSTEM to an OPERABLE status or tripped[a].

   Otherwise, take the ACTION required by Table 3.1.A-1 for the Functional Unit.

3. With one or more CHANNEL(s) required by Table 3.1.A-1 inoperable for Functional Units 13 or 14, within one hour place the inoperable CHANNEL(s) in the tripped condition[a].

   Otherwise, take the ACTION required by Table 3.1.A-1 for the Functional Unit.

4.1 - SURVEILLANCE REQUIREMENTS

A. Reactor Protection System

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 MODE(s) and at the frequencies shown in Table 4.1.A-1.

2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

3. The response time of each reactor trip functional unit shown in Table 3.1.A-1 shall be demonstrated at least once per 18 months. Each test shall include at least one CHANNEL per TRIP SYSTEM such that all CHANNEL(s) are tested at least once every N times 18 months where N is the total number of redundant CHANNEL(s) in a specific reactor TRIP SYSTEM.

---

a. An inoperable CHANNEL or TRIP SYSTEM need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, if the inoperable CHANNEL is not restored to OPERABLE status within the required time, the ACTION required by Table 3.1.A-1 for the Functional Unit shall be taken.

b. This ACTION applies to that TRIP SYSTEM with the most inoperable CHANNELS; if both TRIP SYSTEMS have the same number of inoperable CHANNELS, the ACTION can be applied to either TRIP SYSTEM.

TABLE 3.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATION

(a) When a CHANNEL is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required ACTIONs may be delayed for up to 6 hours provided the Functional Unit maintains RPS trip capability.

(b) This function may be bypassed, provided a control rod block is actuated, for reactor protection system logic reset in Refuel and Shutdown positions of the reactor mode switch.

(c) Deleted.

(d) With THERMAL POWER greater than or equal to 45% of RATED THERMAL POWER.

(e) An APRM CHANNEL is inoperable if there are fewer than 2 LPRM inputs per level or there are less than 50% of the normal complement of LPRM inputs to an APRM CHANNEL.

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.12.A.

(g) Required to be OPERABLE only prior to and during required SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

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

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

TABLE 4.1.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	Applicable OPERATIONAL MODES	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST[r]	CHANNEL[a] CALIBRATION
1. Intermediate Range Monitor:				
a. Neutron Flux - High	2 3, 4, 5	S[b] S	S/U[c], W[o] W[o]	E[o] E[o]
b. Inoperative	2, 3, 4, 5	NA	W[o]	NA
2. Average Power Range Monitor[f]:				
a. Setdown Neutron Flux - High	2 3, 5[m]	S[b] S	S/U[c], W[o] W[o]	SA[o] SA[o]
b. Flow Biased Neutron Flux - High	1	S, D	Q	W[d,e], SA
c. Fixed Neutron Flux - High	1	S	Q	W[d], SA
d. Inoperative	1, 2, 3, 5[m]	NA	Q	NA
3. Reactor Vessel Steam Dome Pressure - High	1, 2[i]	S[q]	Q[q]	E[q][h]
4. Reactor Vessel Water Level - Low	1, 2	D	Q	E[h]
5. Main Steam Line Isolation Valve - Closure	1	NA	Q	E
6. Deleted				
7. Drywell Pressure - High	1, 2[n]	NA	Q	Q

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	Applicable OPERATIONAL MODES	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST[r]	CHANNEL[a] CALIBRATION
8. Scram Discharge Volume Water Level - High				
a. ΔP Switch, and	1, 2, 5[j,k]	NA	Q	E
b. Thermal Switch	1, 2, 5[j,k]	NA	Q	NA
9. Turbine Stop Valve - Closure	1[l]	NA	Q	E
10. Deleted				
11. Turbine Control Valve Fast Closure	1[l]	NA	Q	E
12. Turbine Condenser Vacuum - Low	1	NA	Q	Q
13. Reactor Mode Switch Shutdown Position	1, 2, 3, 4, 5	NA	E	NA
14. Manual Scram	1, 2, 3, 4, 5	NA	M	NA

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least (½) decades during each startup after entering OPERATIONAL MODE 2 and the IRM and APRM channels shall be determined to overlap for at least (½) decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days. The weekly CHANNEL FUNCTIONAL TEST may be used to fulfill this requirement.

(d) This calibration shall consist of the adjustment of the APRM CHANNEL to conform, within 2% of RATED THERMAL POWER, to the power values calculated by a heat balance during OPERATIONAL MODE 1 when THERMAL POWER is ≥25% of RATED THERMAL POWER. This adjustment must be accomplished: a) within 2 hours if the APRM CHANNEL is indicating lower power values than the heat balance, or b) within 12 hours if the APRM CHANNEL is indicating higher power values than the heat balance. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

Any APRM CHANNEL gain adjustment made in compliance with Specification 3.11.B shall not be included in determining the above difference. This calibration is not required when THERMAL POWER is <25% of RATED THERMAL POWER. The provisions of Specification 4.0.D are not applicable.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 2000 effective full power hours (EFPH).

(g) Deleted.

(h) Trip units are calibrated at least once per 92 days and transmitters are calibrated at the frequency identified in the table.

(i) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.12.A.

(j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(k) This function may be bypassed, provided a control rod block is actuated, for reactor protection system reset in Refuel and Shutdown positions of the reactor mode switch.

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

(l) With THERMAL POWER greater than or equal to 45% of RATED THERMAL POWER.

(m) Required to be OPERABLE only prior to and during required SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

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

(o) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for a period of 24 hours after entering OPERATIONAL MODE 2 or 3 when shutting down from OPERATIONAL MODE 1.

(p) Deleted

(q) The CHANNEL CHECK frequency will remain NA, the CHANNEL FUNCTIONAL TEST frequency will remain M, and the CHANNEL CALIBRATION frequency will remain Q for Functional Unit 3 until instrument upgrades are completed (Design Change Package Nos. 9900090 for Unit 1 and 9900091 for Unit 2).

(r) A Functional Test of each Automatic Scram contactor will be performed on a surveillance frequency of W.

BASES

3/4.1.A REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system (RPS) automatically initiates a reactor scram to:

a. preserve the integrity of the fuel cladding,

b. preserve the integrity of the primary system, and

c. minimize the energy which must be absorbed and prevent criticality following a loss-of-coolant accident.

This specification provides the Limiting Conditions for Operation necessary to preserve the ability of the system to perform its intended function, even during periods when instrument CHANNEL(s) may be out-of-service because of maintenance. When necessary, one CHANNEL may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent TRIP SYSTEM(s), each having a minimum of two CHANNEL(s) of tripping devices. Each CHANNEL has an input from at least one instrument CHANNEL which monitors a critical parameter. The outputs of the CHANNEL(s) are combined in a one-out- of-two-logic, i.e., an input signal on either one or both of the CHANNEL(s) will cause a TRIP SYSTEM trip. The outputs of the TRIP SYSTEM(s) are arranged so that a trip on both systems is required to produce a reactor scram. This system meets the intent of the proposed IEEE 279, "Standard for Nuclear Power Plant Protection Systems" issued September 13, 1966. The system has a reliability greater than that of a two-out-of-three system and somewhat less than that of a one-out-of-two system (reference APED 5179). The bases for the trip settings of the RPS are discussed in the Bases for Specification 2.2.A.

Specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies contained in:

General Electric Licensing Topical Report NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.

In order to maintain consistency with the reliability analysis performed in NEDC-30851P-A, the automatic scram contractors will be exercised on a weekly basis. The NEDC-30851P-A analysis concluded that extending surveillance intervals and allowed outage times for RPS instrumentation was acceptable provided the scram contactors were functionally tested on a weekly interval.

The primary reactivity control functions during refueling are the refueling interlocks and the SHUTDOWN MARGIN calculations, which together provide assurance that adequate SHUTDOWN MARGIN is available. The IRMs also provide backup protection for any significant reactivity excursions.

3.2 - LIMITING CONDITIONS FOR OPERATION

A. Isolation Actuation

The isolation actuation instrumentation CHANNEL(s) shown in Table 3.2.A-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

As shown in Table 3.2.A-1.

ACTION:

1. With an isolation actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.A-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

2. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement:

   a) Within 1 hour, verify sufficient CHANNELS remain OPERABLE or in the tripped condition to ensure automatic isolation capability.
   b) Within 12 hours, place the inoperable CHANNEL(s) and/or TRIP SYSTEM in the tripped condition[a] for Table 3.2.A-1 Functional Units common to RPS: 1a, 1b, 2a, 2b, 3a, 3b, 4b, and 7a, and
   c) Within 24 hours, place the inoperable CHANNEL(s) and/or TRIP SYSTEM in the tripped condition[a] for Table 3.2.A-1 Functional Units not common to RPS.

   OR

   Take the ACTION required by Table 3.2.A-1.

4.2 - SURVEILLANCE REQUIREMENTS

A. Isolation Actuation

1. Each isolation actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.2.A-1.

2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

---

a An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, if the inoperable channel is not restored to OPERABLE status within the required time, the ACTION required by Table 3.2.A-1 for the Functional Unit shall be taken.

3.2 - LIMITING CONDITIONS FOR OPERATION

4.2 - SURVEILLANCE REQUIREMENTS

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TABLE 3.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATION

* During CORE ALTERATIONS or operations with a potential for draining the reactor vessel.

** When handling irradiated fuel in the secondary containment.

(a) When a CHANNEL is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required ACTIONs may be delayed for up to 6 hours provided the Functional Unit maintains isolation actuation capability.

(b) Deleted

(c) Isolates the reactor building ventilation system and actuates the standby gas treatment system.

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

(e) Only one TRIP SYSTEM.

(f) Closes only reactor water cleanup system isolation valves.

(g) Only one trip system requied in OPERATIONAL MODE(s) 4 and 5 with RHR Shutdown Cooling System integrity maintained. System integrity is maintained provided the piping is intact and no maintenance is being performed that has the potential for draining the reactor vessel through the system.

(h) Deleted

(i) Includes a time delay of $3 \leq t \leq 9$ seconds.

(j) Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

(k) Also isolates the control room ventilation system.

TABLE 4.2.A-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low	S	Q	E[a]	1, 2, 3
b. Drywell Pressure - High[b]	NA	Q	Q	1, 2, 3
c. Drywell Radiation - High	S	Q	E	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low[c,d]	S	Q	E[a]	1, 2, 3 & *
b. Drywell Pressure - High[b,c,d]	NA	Q	Q	1, 2, 3
c. Reactor Building Ventilation Exhaust Radiation - High[c,d]	S	Q	Q	1, 2, 3 & **
d. Refueling Floor Radiation - High[c,d]	S	Q	Q	1, 2, 3 & **
3. MAIN STEAM LINE (MSL) ISOLATION				
a. Reactor Vessel Water Level - Low Low	S	Q	E[a]	1, 2, 3
b. Deleted				
c. MSL Pressure - Low	NA	Q	Q	1
d. MSL Flow - High[d]	S	Q	E	1, 2, 3
e. MSL Tunnel Temperature - High	NA	E	E	1, 2, 3

TABLE 4.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Standby Liquid Control System Initiation	NA	E	NA	1, 2, 3
b. Reactor Vessel Water Level - Low	S	Q	E[(a)]	1, 2, 3
5. REACTOR CORE ISOLATION COOLING ISOLATION				
a. Steam Flow - High	NA	Q	Q	1, 2, 3
b. Reactor Vessel Pressure - Low	NA	Q	Q	1, 2, 3
c. Area Temperature - High	NA	E	E	1, 2, 3
6. HIGH PRESSURE COOLANT INJECTION ISOLATION				
a. Steam Flow - High	NA	Q	E[(a)]	1, 2, 3
b. Reactor Vessel Pressure - Low	NA	Q	E[(a)]	1, 2, 3
c. Area Temperature - High	NA	E	E	1, 2, 3
7. RHR SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low	S	Q	E[(a)]	3, 4, 5
b. Reactor Vessel Pressure - High (Cut-in Permissive)	NA	Q	Q	1, 2, 3

TABLE 4.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

* During CORE ALTERATIONS or operations with a potential for draining the reactor vessel.

** When handling irradiated fuel in the secondary containment.

(a) Trip units are calibrated at least once per 92 days and transmitters are calibrated at the frequency identified in the table.

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

(c) Isolates the reactor building ventilation system and actuates the standby gas treatment system.

(d) Also isolates the control room ventilation system.

(e) Deleted

3.2 - LIMITING CONDITIONS FOR OPERATION

B. Emergency Core Cooling Systems (ECCS) Actuation

The ECCS actuation instrumentation CHANNEL(s) shown in Table 3.2.B-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

As shown in Table 3.2.B-1.

ACTION:

1. With an ECCS actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.B-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With one or more ECCS actuation instrumentation CHANNEL(s) inoperable, take the ACTION required by Table 3.2.B-1.

4.2 - SURVEILLANCE REQUIREMENTS

B. ECCS Actuation

1. Each ECCS actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.2.B-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

3.2 - LIMITING CONDITIONS FOR OPERATION

4.2 - SURVEILLANCE REQUIREMENTS

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TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[h]	Minimum CHANNEL(s) per Trip Function[a]	Applicable OPERATIONAL MODE(s)	ACTION
3. HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM[d]				
a. Reactor Vessel Water Level - Low Low	≥84 inches	4	1, 2, 3	37
b. Drywell Pressure - High[f]	≤2.5 psig	4	1, 2, 3	37
c. Condensate Storage Tank Level - Low[i]	≥10,000 gal	2	1, 2, 3	35
d. Suppression Chamber Water Level - High[i]	≤14'8" above bottom of chamber	2	1, 2, 3	35
e. Reactor Vessel Water Level - High (Trip)	≤201 inches	2	1, 2, 3	31
f. HPCI Pump Discharge Flow - Low (Bypass)	≥600 gpm	1	1, 2, 3	33
g. Manual Initiation	NA	1/system	1, 2, 3	34
4. AUTOMATIC DEPRESSURIZATION SYSTEM - TRIP SYSTEM 'A' [d]				
a. Reactor Vessel Water Level - Low Low	≥84 inches	2	1, 2, 3	38
b. Drywell Pressure - High[f]	≤2.5 psig	2	1, 2, 3	38
c. Initiation Timer	≤120 sec	1	1, 2, 3	31
d. Low Low Level Timer	≤9.0 min	1	1, 2, 3	31
e. CS Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	2/pump	1, 2, 3	31
f. LPCI Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	2/pump	1, 2, 3	31

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[h]	Minimum CHANNEL(s) per Trip Function[a]	Applicable OPERATIONAL MODE(s)	ACTION
5. AUTOMATIC DEPRESSURIZATION SYSTEM - TRIP SYSTEM 'B' [d]				
a. Reactor Vessel Water Level - Low Low	≥84 inches	2	1, 2, 3	38
b. Drywell Pressure - High[f]	≤2.5 psig	2	1, 2, 3	38
c. Initiation Timer	≤120 sec	1	1, 2, 3	31
d. Low Low Level Timer	≤9.0 min	1	1, 2, 3	31
e. CS Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	2/pump	1, 2, 3	31
f. LPCI Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	2/pump	1, 2, 3	31

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function	Applicable OPERATIONAL MODE(s)	ACTION
6. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	3045±152 volts decreasing voltage	2/bus	1, 2, 3, 4[e], 5[e]	36
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	≥3845 volts (Unit 1)[g][j] ≥3845 volts (Unit 2)[g][j]	2/bus	1, 2, 3, 4[e], 5[e]	36

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

ACTION

ACTION 30 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement (Action 30a only applies in OPERATIONAL MODES 1, 2 and 3):
  a. Within one hour from discovery of loss of initiation capability declare the associated ECCS systems inoperable, AND
  b. Place the inoperable CHANNEL(s) in the tripped condition within 24 hours or declare the associated ECCS system inoperable.

ACTION 31 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:

For CS, LPCI and HPCI (For Functional Units 1.c and 2.c, Action 31a applies only in OPERATIONAL MODES 1, 2, and 3):
  a. Within one hour from discovery of loss of initiation capability declare the associated ECCS systems inoperable, AND
  b. Restore the inoperable CHANNEL(s) to OPERABLE status within 24 hours or declare the associated ECCS system(s) inoperable.

For ADS:
  a. Within one hour from discovery of loss of initiation capability in both ADS trip systems, declare the ADS relief valves inoperable, AND
  b. With RCIC or HPCI inoperable, restore the inoperable CHANNEL(s) to OPERABLE status within 96 hours or declare the ADS relief valves inoperable, AND
  c. With RCIC and HPCI OPERABLE, restore the inoperable CHANNEL(s) to OPERABLE status within 8 days or declare the ADS relief valves inoperable.

ACTION 32 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, place the inoperable CHANNEL in the tripped condition within 24 hours.

ACTION 33 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement (For Functional Units 1.d and 2.d, Action 33a only applies in OPERATIONAL MODES 1, 2 and 3):
  a. Within one hour from discovery of loss of initiation capability declare the associated ECCS system(s) inoperable, AND
  b. Restore the CHANNEL(s) to OPERABLE status within 7 days or declare the associated ECCS system(s) inoperable.

ACTION 34 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, restore the inoperable CHANNEL to OPERABLE status within 24 hours or declare the associated ECCS system(s) inoperable.

ACTION 35 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:
  a. Within one hour from discovery of loss of initiation capability, declare HPCI inoperable, AND
  b. Place the inoperable CHANNEL(s) in the tripped condition within 24 hours or declare the HPCI system inoperable.

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

ACTION

ACTION 36 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, place the inoperable CHANNEL in the tripped condition within one hour, or declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.9.A or 3.9.B, as appropriate.

ACTION 37 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:

a. Within one hour from discovery of loss of initiation capability declare HPCI inoperable, AND
b. Place the inoperable CHANNEL(s) in the tripped condition within 24 hours or declare HPCI inoperable.

ACTION 38 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:

a. Within one hour from discovery of loss of initiation capability in both ADS trip systems, declare the ADS relief valves inoperable, AND
b. With RCIC or HPCI inoperable, place the inoperable CHANNEL(s) in the tripped condition within 96 hours or declare the ADS relief valves inoperable, AND
c. Place the inoperable CHANNEL(s) in the tripped condition within 8 days or, declare the ADS relief valves inoperable.

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

TABLE NOTATION

(a) When a CHANNEL is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required ACTIONs may be delayed as follows:

1) For up to six hours for Functional Units 3.e, 3.f, and 3.g; and
2) For up to six hours for Functional Units other than 3.e, 3.f, and 3.g provided the functional unit maintains actuation capability.

(b) Also actuates the associated emergency diesel generator.

(c) When the system is required to be OPERABLE per Specification 3.5.B.

(d) Not required to be OPERABLE when reactor steam dome pressure is ≤150 psig.

(e) Required when the associated diesel generator is required to be OPERABLE per Specification 3.9.B.

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

(g) With no LOCA signal present, there is an additional time delay of 5±0.25 minutes.

(h) Reactor water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

(i) Provides signal to pump suction valves only.

(j) There is an inherent time delay of 7 ± 1.4 seconds on degraded voltage.

TABLE 4.2.B-1

ECCS ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. CORE SPRAY (CS) SYSTEM				
a. Reactor Vessel Water Level - Low Low	S	M	BIM	1, 2, 3, 4(b), 5(b)
b. Drywell Pressure - High(d)	NA	Q	Q	1, 2, 3
c. Reactor Vessel Pressure - Low (Permissive)	NA	Q	Q	1, 2, 3, 4(b), 5(b)
d. CS Pump Discharge Flow - Low (Bypass)	NA	Q	E(e)	1, 2, 3, 4(b), 5(b)
2. LOW PRESSURE COOLANT INJECTION (LPCI) SUBSYSTEM				
a. Reactor Vessel Water Level - Low Low	S	M	BIM	1, 2, 3, 4(b), 5(b)
b. Drywell Pressure - High(d)	NA	Q	Q	1, 2, 3
c. Reactor Vessel Pressure - Low (Permissive)	NA	Q	Q	1, 2, 3, 4(b), 5(b)
d. LPCI Pump Discharge Flow - Low (Bypass)	NA	Q	E(e)	1, 2, 3, 4(b), 5(b)
3. HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM(a)				
a. Reactor Vessel Water Level - Low Low	S	M	BIM	1, 2, 3
b. Drywell Pressure - High(d)	NA	Q	Q	1, 2, 3
c. Condensate Storage Tank Level - Low	NA	Q	NA	1, 2, 3
d. Suppression Chamber Water Level - High	NA	Q	NA	1, 2, 3
e. Reactor Vessel Water Level - High (Trip)	NA	M	BIM	1, 2, 3
f. HPCI Pump Discharge Flow - Low (Bypass)	NA	Q	E	1, 2, 3
g. Manual Initiation	NA	E	NA	1, 2, 3

TABLE 4.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
4. AUTOMATIC DEPRESSURIZATION SYSTEM[a]				
a. Reactor Vessel Water Level - Low Low	S	M	BIM	1, 2, 3
b. Drywell Pressure - High[d]	NA	Q	Q	1, 2, 3
c. Initiation Timer	NA	E	E	1, 2, 3
d. Low Low Level Timer	NA	E	E	1, 2, 3
e. CS Pump Discharge Pressure - High (Permissive)	NA	Q	Q	1, 2, 3
f. LPCI Pump Discharge Pressure - High (Permissive)	NA	Q	Q	1, 2, 3
5. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	NA	E	E	1, 2, 3, 4[c], 5[c]
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	NA	E	E	1, 2, 3, 4[c], 5[c]

TABLE 4.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Not required to be OPERABLE when reactor steam dome pressure is ≤150 psig.

(b) When the system is required to be OPERABLE per Specification 3.5.B.

(c) Required when the associated diesel generator is required to be OPERABLE per Specification 3.9.B.

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

(e) Trip units are calibrated at least once per 92 days and transmitters are calibrated at the frequency identified in the table.

TABLE 3.2.C-1

ATWS - RPT INSTRUMENTATION

Functional Unit	Trip Setpoint[c]	Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM[a]
1. Reactor Vessel Water Level - Low Low	≥84 inches[b]	2
2. Reactor Vessel Pressure - High	≤1250 psig	2

---

a When a CHANNEL is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required ACTIONs may be delayed for up to 6 hours provided the Functional Unit maintains ATWS actuation capability

b Includes a time delay of $8 \leq t \leq 10$ seconds.

c Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

TABLE 4.2.C-1

ATWS - RPT INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Water Level - Low Low	S	Q	E[(a)]
2. Reactor Vessel Pressure - High	S	Q	E[(a)]

a Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

TABLE 3.2.D-1

REACTOR CORE ISOLATION COOLING ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[c]	Minimum CHANNEL(s) per Trip Function[a]	ACTION
1. Reactor Vessel Water Level - Low Low	≥84 inches	4	40
2. Reactor Vessel Level - High (Trip)	≤201 inches	2	41
3. Condensate Storage Tank Level - Low	≥598' El.	2[b]	42
4. Suppression Chamber Water Level - High	≤14'8" above bottom of chamber	2[b]	42
5. Manual Initiation	NA	1/system	43

---

a When a CHANNEL is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required ACTIONs may be delayed as follows:

1) For up to six hours for Functional Units 2 and 5; and
2) For up to six hours for Functional Units other than 2 and 5 provided the functional unit maintains actuation capability.

b. Provides signal to pump suction valves only.

c Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

TABLE 3.2.D-1(Continued)

REACTOR CORE ISOLATION COOLING ACTUATION INSTRUMENTATION

ACTION

ACTION 40- With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:

a. Within one hour from discovery of loss of initiation capability declare RCIC inoperable, AND
b. Place the inoperable CHANNEL(s) in the tripped condition within 24 hours or declare RCIC inoperable.

ACTION 41 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement, declare the RCIC system inoperable within 24 hours.

ACTION 42 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:

a. Within one hour from discovery of loss of initiation capability declare RCIC inoperable, AND
b. Place the inoperable CHANNEL(s) in the tripped condition within 24 hours or declare RCIC inoperable.

ACTION 43 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement, restore the inoperable CHANNEL to OPERABLE status within 24 hours or declare the RCIC system inoperable.

TABLE 4.2.D-1

REACTOR CORE ISOLATION COOLING ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low Low	S	M	BIM
2. Reactor Vessel Water Level - High (Trip)	S	M	BIM
3. Condensate Storage Tank Level - Low	NA	Q	NA
4. Suppression Chamber Water Level - High	NA	Q	NA
5. Manual Initiation	NA	E	NA

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION

ACTION 50 - Declare the rod block monitor inoperable and take the ACTION required by Specification 3.3.M.

ACTION 51- With the number of OPERABLE CHANNEL(s):

a. One less than required by the Minimum CHANNEL(s) 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 CHANNEL(s) per Trip Function requirement, place at least one inoperable CHANNEL in the tripped condition within one hour.

ACTION 52 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, place the inoperable CHANNEL in the tripped condition within 12 hours.

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

TABLE NOTATION

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

(b) This function shall be automatically bypassed if the IRM channels are on range 3 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 1.

(e) With THERMAL POWER ≥30% of RATED THERMAL POWER.

(f) With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(g) 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 3.11.B. W is equal to the percentage of the drive flow required to produce a rated core flow of 98 x 10^6 lbs/hr.

(h) Required to be OPERABLE only during SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(i) When a CHANNEL is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required ACTIONs may be delayed for up to 6 hours provided the Functional Unit maintains rod block actuation capability.

(j) With detector count rate less than or equal to 100 cps.

TABLE 4.2.E-1

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
1. ROD BLOCK MONITORS				
a. Upscale	NA	S/U[b,c], Q[c]	Q	1[d]
b. Inoperative	NA	S/U[b,c], Q[c]	NA	1[d]
c. Downscale	NA	S/U[b,c], Q[c]	Q	1[d]
2. AVERAGE POWER RANGE MONITORS				
a. Flow Biased Neutron Flux - High				
1. Dual Recirculation Loop Operation	NA	S/U[b], Q	SA	1
2. Single Recirculation Loop Operation	NA	S/U[b], Q	SA	1
b. Inoperative	NA	S/U[b], Q	NA	1, 2, 5[j]
c. Downscale	NA	S/U[b], Q	SA	1
d. Startup Neutron Flux - High	NA	S/U[b], Q	SA	2, 5[j]
3. SOURCE RANGE MONITORS				
a. Detector not full in[f]	NA	S/U[b], W	E	2[i][k], 5[k]
b. Upscale[g]	NA	S/U[b], W	E	2[i], 5
c. Inoperative[g]	NA	S/U[b], W	NA	2[i], 5

TABLE 3.2.I-1

SUPPRESSION CHAMBER AND DRYWELL SPRAY ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[a]	Minimum CHANNEL(s) per TRIP SYSTEM[c]	ACTION
1. Drywell Pressure - High (Permissive)	$0.5 \leq \rho \leq 1.5$ psig	2	80
2. Reactor Vessel Water Level - Low (Permissive)	$\geq$ -48 inches	1	80

ACTION

ACTION 80 -

a. With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place at least one inoperable CHANNEL in the tripped condition[b] within 24 hours or declare the Suppression Chamber and Drywell Spray Actuation mode of the Residual Heat Removal system inoperable.

b. With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for both TRIP SYSTEM(s), declare the Suppression Chamber and Drywell Spray Actuation mode of the Residual Heat Removal system inoperable.

---

a Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

b If an instrument is inoperable, it shall be placed (or simulated) in a tripped condition so that it will not prevent a containment spray.

c When a CHANNEL is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required ACTIONs may be delayed for up to 6 hours provided the Functional Unit maintains Suppression Chamber and Drywell Spray actuation capability.

TABLE 4.2.I-1

SUPPRESSION CHAMBER AND DRYWELL SPRAY ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Drywell Pressure - (Permissive)	NA	Q	Q
2. Reactor Vessel Water Level - Low (Permissive)	D	Q	E[a]

a Trip units are calibrated at least once per 92 days and transmitters are calibrated at the frequency indicated in the table.

BASES

3/4.2 INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram (Sections 2.2 and 3/4.1), protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or which terminates operator errors before they result in serious consequences. The objectives of these specifications are to assure the effectiveness of the protective instrumentation when required and to prescribe the trip settings required to assure adequate performance. As indicated, one CHANNEL may be required to be made inoperable for brief intervals to conduct required surveillance. Some of the settings have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations. Surveillance requirements for the instrumentation are selected in order to demonstrate proper function and OPERABILITY. Additional instrumentation for REFUELING operations is identified in Sections 3/4.10.B.

Current fuel designs incorporate slight variations in the length of the active fuel, and thus the actual top of active fuel, when compared with the original fuel designs. Safety Limits, instrument water level setpoints, and associated LCOs refer to the top of active fuel. In these cases, the top of active fuel is defined as 360 inches above vessel zero. Licensing analyses, both accident and transient, utilize this definition for the automatic initiation and manual intervention associated with these events.

3/4.2.A Isolation Actuation Instrumentation

The isolation actuation instrumentation automatically initiates closure of appropriate isolation valves and/or dampers, which are necessary to prevent or limit the release of fission products from the reactor coolant system, the primary containment and the secondary containment in the event of a loss-of-coolant accident or other reactor coolant pressure boundary (RCPB) leak. The parameters which result in isolation of the secondary containment also actuate the standby gas treatment system. The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary and secondary containment and RCPB system isolation. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. Redundant sensor input signals for each parameter are provided for initiation of isolation (one exception is standby liquid control system initiation).

The reactor low level instrumentation is set to trip at greater than or equal to 144 inches above the top of active fuel (which is defined to be 360 inches above vessel zero). This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps. For this trip setting and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs, even for the maximum break.

Specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies contained in:

BASES

General Electric Licensing Topical Report, NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

General Electric Licensing Topical Report, NEDC 31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

3/4.2.B Emergency Core Cooling System Actuation Instrumentation

The emergency core cooling system (ECCS) instrumentation generates signals to automatically actuate those safety systems which provide adequate core cooling in the event of a design basis transient or accident. The instrumentation which actuates the ECCS is generally arranged in a one-out-of-two taken twice logic circuit. The logic circuit is composed of four CHANNEL(s) and each CHANNEL contains the logic from the functional unit sensor up to and including all relays which actuate upon a signal from that sensor. For core spray and low pressure coolant injection, the divisionally powered actuation logic is duplicated and the redundant components are powered from the other division's power supply. The single-failure criterion is met through provisions for redundant core cooling functions, e.g., sprays and automatic blowdown and high pressure coolant injection. Although the instruments are listed by system, in some cases the same instrument is used to send the actuation signal to more than one system at the same time.

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

Specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies contained in:

General Electric Licensing Topical Report, NEDC-30936P-A, Part 1 and Part 2, "Technical Specification Improvement Methodology With Demonstration for BWR ECCS Actuation Instrumentation," December 1988.

3/4.2.C ATWS - RPT Instrumentation

The anticipated transient without scram (ATWS) recirculation pump trip (RPT) provides a means of limiting the consequences of the unlikely occurrence of a failure to scram concurrent with the associated anticipated transient. The response of this plant to this postulated event falls within the bounds of study events in General Electric Company Topical Report NEDO-10349, dated March 1971

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and NEDO24222, dated December 1979. Tripping the recirculation pumps adds negative reactivity by increasing steam voiding in the core area as core flow decreases.
Specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies contained in:

General Electric Licensing Topical Report, GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.

3/4.2.D Reactor Core Isolation Cooling Actuation Instrumentation

The reactor core isolation cooling system provides makeup water to the core in the event of a postulated isolation of the reactor from the main condenser with a loss of feedwater. The system automatically initiates upon receipt of a reactor vessel low-low water level signal utilizing level indicating switches in a one-out-of-two taken twice logic scheme. The system may also be manually started.

Specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies contained in:

General Electric Licensing Topical Report, GENE 770-06-2-A, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.

3/4.2.E Control Rod Block Actuation Instrumentation

The control rod block functions are provided to prevent excessive control rod withdrawal so that the MINIMUM CRITICAL POWER RATIO (MCPR) does not go below the MCPR fuel cladding integrity Safety Limit. During shutdown conditions, control rod block instrumentation initiates withdrawal blocks to ensure that all control rods remain inserted to prevent inadvertent criticality.

The trip logic for this function is one-out-of-n; e.g., any trip on one of the six average power range monitors (APRMs), eight intermediate range monitors (IRMs), or four source range monitors (SRMs), will result in a rod block. The minimum instrument CHANNEL requirements assure sufficient instrumentation to assure that the single failure criterion is met. The minimum instrument CHANNEL requirements for the rod block monitor may be reduced by one for a short period of time to allow for maintenance, testing, or calibration.

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

In the REFUEL MODE during SHUTDOWN MARGIN demonstrations and the STARTUP/HOT STANDBY OPERATIONAL MODE, the APRM rod block function setpoint is significantly reduced to provide the same type of protection in the REFUEL and STARTUP/HOT STANDBY OPERATIONAL

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MODE(s) as the APRM flow-biased rod block does in the RUN OPERATIONAL MODE, i.e., prevents control rod withdrawal before a scram is reached.

The rod block monitor (RBM) function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error. The trip setting is flow-biased. At low power, the worst-case withdrawal of a single control rod without rod block action will not violate the fuel cladding integrity Safety Limit. Thus the RBM rod block function is not required below the specified power level. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the fuel cladding integrity Safety Limit.

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

A downscale indication on an APRM is an indication that the instrument has failed or is not sufficiently sensitive. In either case, the instrument will not respond to changes in control rod motion, and the control rod motion is thus prevented.

The SRM rod blocks of low count rate and the detector not fully inserted assure that the SRMs are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram discharge volume, high water level rod block provides annunciation for operator action. The alarm setpoint has been selected to provide adequate time to allow for the determination of the cause for the level increase and corrective action prior to automatic scram initiation.

Specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies contained in:

General Electric Licensing Topical Report, GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.

General Electric Licensing Topical Report, NEDC-30851P-A, Supplement 1, "Technical Specification Improvement Analysis For BWR Control Rod Block Instrumentation," October 1988.

3/4.2.F Accident Monitoring Instrumentation

Instrumentation is provided to monitor sufficient accident conditions to adequately assess important variables and provide operators with necessary information to complete the appropriate mitigation actions. OPERABILITY of the instrumentation listed provides adequate monitoring of the containment following a loss-of-coolant accident. Information from this instrumentation will provide the operator with a detailed knowledge of the conditions resulting from the accident; based on this information, the operator can make logical decisions regarding post accident recovery. Allowable outage times are based on diverse instrumentation availability for guiding the operator should an accident occur, and on

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the low probability of an instrument being out-of-service concurrent with an accident. As noted in the surveillance requirements, the instrumentation CHANNEL(s) associated with Torus Pressure provides a dual function and is shared in common with the Drywell Pressure (Narrow and Wide Ranges) instrumentation. This instrumentation is identified in response to Generic Letter 82-33 and the associated NRC Safety Evaluation Report, and some instrumentation is included in accordance with the response to Generic Letter 83-36.

3/4.2.G Source Range Monitoring Instrumentation

The source range monitors (SRM) provide the operator with the status of the neutron flux in the core at very low power levels during startup and shutdown. The consequences of reactivity accidents are functions of the initial neutron flux. Therefore, the requirements for a minimum count rate assures that any transient, should it occur, begins at or above the initial value used in the analyses of transients from cold conditions. Two OPERABLE SRM CHANNEL(s) are adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. Three OPERABLE SRMs provide an added conservatism. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.2.H Explosive Gas Monitoring Instrumentation

Instrumentation is provided to monitor the concentrations of potentially explosive mixtures in the off-gas holdup system to prevent a possible uncontrolled release via this pathway. This instrumentation is included in accordance with Generic Letter 89-01.

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3/4.2.I Suppression Chamber and Drywell Spray Actuation Instrumentation

Instrumentation is provided to monitor the parameters which are necessary to permit initiation of the containment cooling mode of the residual heat removal system to condense steam in the containment atmosphere. The spray mode does not significantly affect the rise of drywell pressure following a loss of coolant accident, but does result in quicker depressurization following completion of the blowdown.

Specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies contained in:

General Electric Licensing Topical Report, GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.

3/4.2.J Feedwater Trip System Actuation

The feedwater trip system actuation instrumentation is designed to detect a potential failure of the feedwater control system which causes excessive feedwater flow. If undetected, this would lead to reactor vessel water carryover into the main steam lines and to the main turbine. This instrumentation is included in response to Generic Letter 89-19.

3/4.2.K Toxic Gas Monitoring

Toxic gas monitoring instrumentation is provided in or near the control room ventilation system intakes to allow prompt detection and the necessary protective actions to be initiated. Isolation from high toxic chemical concentration has been added to the station design as a result of the "Control Room Habitability Study" submitted to the NRC in December 1981 in response to NUREG-0737 Item III D.3.4. As explained in Section 3 of this study, ammonia, chlorine, and sulphur dioxide detection capability has been provided. In a report generated by Sargent and Lundy in April 1991, justification was provided to delete the chlorine and sulphur dioxide detectors from the plant. The setpoints chosen for the control room ventilation isolation are based on early detection in the outside air supply at the odor threshold, so that the toxic chemical will not achieve toxicity limit concentrations in the Control Room.

3/4.2.L Mechanical Vacuum Pump Isolation Instrumentation

The Mechanical Vacuum Pump Isolation Instrumentation initiates a trip of the main condenser Mechanical Vacuum Pump following an event in which main steam line radiation levels exceed predetermined values. Tripping the mechanical vacuum pump limits the offsite doses in the event of a control rod drop accident (CRDA). The trip logic consists of two independent trip systems with two channels of the Main Steam Line Radiation - High in each trip system. The outputs of each trip system are combined in a one-out-of-two taken twice logic.

The trip of the Mechanical Vacuum Pump is credited in the CRDA radiological analysis. Accordingly, the Mechanical Vacuum Pump trip is required to be operable in Modes 1 and 2. In modes 3, 4 and 5,

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the consequences of a CRDA are insignificant and are not expected to result in any fuel damage. Surveillance requirements for testing and calibration are provided to ensure an acceptable level of quality and reliability.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 198 TO FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. 194 TO FACILITY OPERATING LICENSE NO. DPR-30

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-254 AND 50-265

1.0 INTRODUCTION

By a letter dated December 27, 1999, Exelon Generation Company, LLC (EGC, or the licensee, formerly Commonwealth Edison Company), the licensee for Quad Cities Nuclear Power Station, Units 1 and 2, requested NRC's approval to amend its operating licenses DPR-29 and DPR-30 by revising the plant's Technical Specifications (TS). The proposed amendment will increase surveillance test intervals (STIs) and allowed outage times (AOTs) for selected TS instrumentation and will revise the associated Bases sections to reflect the increased STIs and AOTs. A few administrative changes are also being proposed to support the conversion to the Improved Standard Technical Specifications (ISTS). In its submittal, the licensee stated that the proposed changes are consistent with the instrument STIs and AOTs found in the ISTS, NUREG-1433, Revision 1, "Standard Technical Specifications, General Electric Plants, BWR/4," and similar changes were approved by the staff for the Brunswick Steam Electric Plant, Units 1 and 2, in the issuance of operating license amendment No. 175, dated March 30, 1995. The proposed changes at Quad Cities are expected to be implemented in the near future to support conversion to the ISTS.

2.0 BACKGROUND

In 1983, the Boiling Water Reactor (BWR) Owner's Group (BWROG) formed a Technical Specifications Improvement (TSI) Committee. This committee established a program to identify improvements to AOTs and STIs specified in BWR standard TS. The primary objective of this program was to minimize unnecessary testing and restrictive AOTs that could degrade overall safety and availability of equipment. During April 1984, the TSI committee met with the NRC and outlined the BWR TSI program. The NRC expressed agreement with the overall approach. Subsequently, the BWROG developed a series of General Electric (GE) licensing topical reports (LTRs) that provided the basis for extending STIs and AOTs for key systems' actuation instrumentation, including the reactor protection system (RPS), the emergency core

cooling system (ECCS), the containment isolation system, control rod block functions, and other miscellaneous safety functions. These LTRs were reviewed and approved by the staff. The staff's approval of these LTRs included requirements that plant-specific applications confirm the applicability of the generic analyses to the specific plant and also that the setpoint drift, which could be expected under the extended test intervals, be within the existing allowances in the respective instrument setpoint calculations.

Proposed Changes: The proposed changes will permit channel functional tests to be conducted quarterly rather than weekly or monthly in accordance with the current schedule, and the AOTs will be increased from the current 1 hour to 12 or 24 hours for repairs, and from 2 hours to 6 hours for surveillance testing. Attachment B to the licensee's submittal dated December 27, 1999, includes marked-up pages of the current TS and Bases associated with the proposed changes. Attachment E to the licensee's submittal provided site specific evaluations (Refs. 10, 11, and 15) to support the proposed changes. The licensee's submittal describes the proposed changes and references supporting documentation for the changes as follows:

TS Section 1.0 - Definitions:

Change No.	Description of Change	Reference(s)
1)	Page 1-8: Surveillance Frequency Notation BIM, equal to 60 days, has been added to Table 1-1 of TS Section 1.0, Definitions. The new frequency definition will be used on certain instruments as detailed in Change Nos. 24 and 34 below.	None

TS Section 3/4.1.A - Reactor Protection System (RPS):

Change No.	Description of Change	Reference(s)
2)	Page 3/4.1-1: Revise Actions 1 and 2 and Footnotes a and b to incorporate a 1 hour check for trip capability (loss-of-function) and to provide a 12 hour AOT for maintenance activities. Note that the manual scram functions (Functional Units 13 and 14) were not changed to allow a 12 hour AOT because their configuration is not consistent with the generic model evaluated in Reference 3.	2,3
3)	Page 3/4.1-6: Revise Table 3.1.A-1, Note (a), to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability consistent with ISTS.	1,3

Change No.	Description of Change	Reference(s)
4)	Page 3/4.1-7: Add Note (r) to Table 4.1.A-1 to provide for weekly testing of automatic scram contactors. As discussed in Reference 3, sensor channel tests can be increased to quarterly provided the automatic scram contactors are tested on a weekly basis.	3
5)	Pages 3/4.1-7, 3/4.1-8 and 3/4.1-9: Revise Table 4.1.A-1 to extend the CHANNEL FUNCTIONAL TEST frequency to quarterly for the following Functional Units: 2.b, 2.c, 2.d, 3, 4, 5, 6, 7, 9, 10, 11 and 12. Note (q) to TS Table 4.1.A-1 has been added on page 3/4.1-10 to specifically address planned upgrades to Functional Unit 3. Upgrades are planned for the upcoming Unit 2 and Unit 1 refueling outages for Functional Unit 3 that will support a CHANNEL FUNCTIONAL TEST frequency of Q and CHANNEL CALIBRATION frequency of E. A shiftly (S) CHANNEL check has also been provided consistent with ISTS. Table 4.1.A-1, Note (h), has been changed to reflect a 92-day calibration of associated trip units.	3
6)	Page B 3/4.1-1: Bases changes indicating that specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies.	3

TS Section 3/4.2.A – Isolation Actuation:

Change No.	Description of Change	Reference(s)
7)	Pages 3/4.2-1 and 3/4.2-2: Revise Action 2 to incorporate a 1 hour check for loss-of-function and to provide AOTs of 12 hours to repair Functional Units that are common to RPS and 24 hours to repair Functional Units that are not common to RPS. Due to loss-of-function check provided in Action 2, Action 3 has been deleted. Due to the deletion of Action 3, Footnotes (b) and (c) have been deleted. Revise Footnote (a) to reflect change in AOT requirement.	4,5,13,11
8)	Page 3/4.2-7: Revise Table 3.2.A-1, Note (a), to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability consistent with ISTS.	1,4,5,11
9)	Pages 3/4.2-8, 3/4.2-9 and 3/4.2-10: Revise Table 4.2.A-1 to increase the Channel Functional Test intervals from M to Q for the following Functional Units: 1.a, 1.b, 1.c, 2.a, 2.b, 2.c, 2.d, 3.a, 3.b, 3.c, 3.d, 4.b, 5.a, 5.b, 6.a, 6.b, 7.a and 7.b. Table 4.2.A-1, Note (a), has been modified to reflect the increase in the Channel Calibration interval for the corresponding Trip Units from 31 to 92 days.	4,5,11
10)	Page B 3/4.2-1: Bases change indicating that specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies.	4,5

TS Section 3/4.2.B – Emergency Core Cooling System (ECCS) Actuation:

Change No.	Description of Change	Reference(s)
11)	Pages 3/4.2-11 and 3/4.2-12: Action 3 has been deleted. The Action 3 requirements have been incorporated into a new Action 38 for ADS initiation instrumentation. This change is consistent with ISTS.	1

Change No.	Description of Change	Reference(s)
12)	Pages 3/4.2-14 and 3/4.2-15: For ADS permissive functions 4e, 4f, 5e and 5f, the Minimum Channels per Trip Function requirement has been increased from 1/pump to 2/pump. This more restrictive change ensures each ADS trip system has a sufficient number of operable channels to initiate during a design basis event. This change is consistent with the ISTS.	1
13)	Page 3/4.2-14: A new Action 37 is proposed for HPCI Initiation Functional Units 3a and 3b. This conservative change ensures HPCI injection capability and is consistent with the ISTS.	1
14)	Pages 3/4.2-14 and 3/4.2-15: A new Action 38 has been incorporated for ADS initiation functions 4a, 4b, 5a, and 5b. This change is consistent with ISTS.	1
15)	Page 3/4.2-16: Revise Action 30 to incorporate a 1 hour loss-of-function check and a 24 hour AOT for maintenance activities. Action 30a is only applicable in Modes 1, 2 and 3 because in Modes 4 and 5 the specific initiation time of low pressure ECCS is not assumed and the probability of a LOCA is lower. This change is consistent with the ISTS.	1,6,14,15
16)	Page 3/4.2-16: Revise Action 31 to incorporate a 1 hour loss-of-function check and a 24 hour AOT for maintenance activities. For CS, LPCI and HPCI, Action 31a is only applicable in Modes 1, 2 and 3 because in Modes 4 and 5 the specific initiation time of ECCS is not assumed and the probability of a LOCA is lower. This change is consistent with the ISTS.	1,6,14,15
17)	Page 3/4.2-16: Revise Action 32 to extend the AOT for maintenance activities to 24 hours.	6,14,15
18)	Page 3/4.2-16: Revise Action 33 to incorporate a 1 hour loss-of-function check. For Functional Units 1.d and 2.d, Action 33a is only applicable in Modes 1, 2 and 3 because in Modes 4 and 5 the specific initiation time of low pressure ECCS is not assumed and the probability of a LOCA is lower. This change is consistent with the ISTS.	1
19)	Page 3/4.2-16: Revise Action 34 to extend the AOT for maintenance activities to 24 hours.	6,14,15

Change No.	Description of Change	Reference(s)
20)	Page 3/4.2-16: Revise Action 35 to incorporate a 1 hour loss-of-function check and a 24 hour AOT for maintenance activities. This change is consistent with the ISTS.	1,6,14,15
21)	Page 3/4.2-16: A New Action 37 is proposed for HPCI initiation functions 3a and 3b. This change incorporates a 1 hour loss-of-function check and a revised AOT for maintenance activities. This change is consistent with ISTS.	1,6,14,15
22)	Page 3/4.2-16: A new Action 38 is proposed for ADS initiation functions 4a, 4b, 5a, and 5b. This new action incorporates a 1 hour loss-of-function check and a revised AOT for maintenance activities. If the action requirements can not be met, the ADS relief valves are declared inoperable. The actions for inoperable ADS relief valves are provided in TS 3.6.F, which provides ACTION requirements consistent with 3.2.B, Action 3 (which has been deleted by this proposed change).	1,6,14,15
23)	Page 3/4.2-17: Revise Table 3.2.B-1, Note (a), to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability. These changes are consistent with the ISTS and the reliability analysis performed in Reference 6.	1,6,15
24)	Pages 3/4.2-18, 3/4.2-19, and 3/4.1-20: Revise Table 4.2.B-1 to increase the Channel Functional Test intervals from M to Q for the following Functional Units: 1.b, 1.c, 1.d, 2.b, 2.c, 2.d, 3.b, 3.c, 3.d, 3.f, 4.b, 4.e and 4.f. The Channel Calibration intervals for Functional Units 1.a, 2.a, 3.a, 3.e, and 4.a have been changed to BIM to reflect the current station practices. Table 4.2.B-1, Note (e), has been changed to reflect a 92-day calibration of the associated trip units.	6,15
25)	Page B 3/4.2-2: Bases changes indicating that specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies.	6

TS Section 3/4.2.C – Anticipated Transient Without Scram (ATWS) - Recirculation Pump Trip (RPT):

Change No.	Description of Change	Reference(s)
26)	Page 3/4.2-23: Revise Table 3.2.C-1, Note (a), to allow a 6-hour AOT for testing and to incorporate a check for trip capability consistent with ISTS.	1,7
27)	Page 3/4.2-24: Revise Table 4.2.C-1 to increase the Channel Functional Test intervals from M to Q for the following Functional Units: 1 and 2.	7
28)	Page B 3/4.2-2: Bases changes indicating that specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies.	7

TS Section 3/4.2.D – Reactor Core Isolation Cooling (RCIC) Actuation Instrumentation:

Change No.	Description of Change	Reference(s)
29)	Page 3/4.2-26: Revise Table 3.2.D-1, Note (a), to allow a 6-hour AOT for testing and to incorporate a check for trip capability. These changes are consistent with the ISTS and the reliability analysis performed in Reference 8.	1,8
30)	Page 3/4.2-27: Revise Action 40 to incorporate a 1 hour loss-of-function check and a 24 hour AOT for maintenance activities. This change is consistent with the Improved Standard Technical Specifications.	1,8
31)	Page 3/4.2-27: Change Action 41 to allow 24 AOT for maintenance activities.	8
32)	Page 3/4.2-27: Revise Action 42 to incorporate a 1 hour loss-of-function check and a 24 hour AOT for maintenance activities. This change is consistent with ISTS.	1,8
33)	Page 3/4.2-27: Change Action 43 to allow 24 AOT for maintenance activities.	8
34)	Page 3/4.2-28: Revise Table 4.2.D-1 to increase the Channel Functional Test intervals from M to Q for the following Functional Units: 3 and 4. The Channel Calibration intervals for Functional Units 1 and 2 have been changed to BIM to reflect the current station practices.	8

Change No.	Description of Change	Reference(s)
35)	Page B 3/4.2-2: Bases changes indicating that specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies.	8

TS Section 3/4.2.E – Control Rod Block Actuation:

Change No.	Description of Change	Reference(s)
36)	Page 3/4.2-33: Revise Table 3.2.E-1, Action 52, to increase the AOT for maintenance activities from one hour to 12 hours.	7
37)	Page 3/4.2-34: Revise Table 3.2.E- 1, Note (i), to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability consistent with ISTS.	1,7
38)	Page 3/4.2-35: Revise Table 4.2.E-1 to increase the Channel Functional Test intervals from M to Q for the following Functional Units: 1.a, 1.b, 1.c, 2.a.1, 2.a.2, 2.b, 2.c and 2.d.	9
39)	Page B 3/4.2-4: Bases changes indicating that specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies.	7,9

TS Section 3/4.2.I – Suppression Chamber and Drywell Spray Actuation:

Change No.	Description of Change	Reference(s)
40)	Page 3/4.2-49: Revise Table 3.2.I-1, Action 80.a, to increase the AOT for maintenance activities from one hour to 24 hours.	7
41)	Page 3/4.2-49: Revise Table 3.2.I-1, Note (c), to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability consistent with ISTS.	1,7

Change No.	Description of Change	Reference(s)
42)	Page 3/4.2-50: Revise Table 4.2.I-1 to increase the Channel Functional Test interval from M to Q for the following Functional Units: 1 and 2. In addition, Table 4.2.I-1, Note (a), has been changed to reflect a 92-day calibration of the associated trip units.	7
43)	Page B 3/4.2-5: Bases changes indicating that specified surveillance intervals and surveillance and maintenance allowable outage times have been determined in accordance with the NRC approved methodologies.	7

In addition, station administrative procedures are being revised to increase: (1) the test frequency of scram contactors, and (2) the calibration frequency of steam dome high pressure and the reactor low-low level instruments and the proposed STI and AOT extension is not requested for these instruments. All associated Bases sections will be revised to reflect these proposed changes and to indicate that specified surveillance intervals and surveillance and maintenance AOTs have been determined in accordance with the NRC-approved LTRs.

3.0 EVALUATION

In their submittal, the licensee stated that all proposed changes are consistent with the ISTSs and the LTRs, which have been previously approved by the staff. The LTRs assessed the reliability of TS instrumentation and concluded that extending STIs and AOTs for testing and repair enhances operational safety because: (1) the potential for inadvertent plant scrams will be reduced; (2) the number of test cycles on equipment will be minimized; (3) unwarranted radiation exposure to plant personnel can be reduced; and (4) the use of plant personnel can be optimized.

While approving the LTRs, the staff stipulated that licensee submittals for extending STIs and AOTs in accordance with the LTRs shall: (1) confirm the applicability of the generic analyses to their specific plant; (2) demonstrate, by use of current drift information, that setpoint drift of all the associated instruments under the extended test intervals are bounded by the assumptions used in the generic analyses and will be within the existing allowances in their respective instrument setpoint calculations; and (3) confirm that evaluation for the differences between the parts of the RPS that perform the trip functions in the specific plant and those of the base case-specific plant analysis included in the generic analysis of the LTR, has been performed in accordance with procedures in Appendix K of NEDC-30851P.

In their submittal dated December 27, 1999, the licensee confirmed the applicability of generic analyses to Quad Cities, Units 1 and 2, and also confirmed that the expected setpoint drift for all affected instruments was found to be within the existing allowances assumed in the respective instrument setpoint calculations. The licensee further stated that the analysis performed to identify and evaluate the differences between the parts of

the RPS that perform the trip functions at Quad Cities and those analyzed in the generic study utilized the methodology outlined in Appendix K of LTR NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988. The licensee further added that its site-specific analysis indicated that although the RPS configurations for Quad Cities differ in some respects from the configuration of the base plant analyzed in the LTR, the differences do not have a significant impact on the results and conclusions of the LTR generic evaluation. The base case indicates that the increase in the frequency of the scram contactor testing creates a low probability of scram failure. Therefore, as required by LTR NEDC-30851P-A, the proposed TS changes include a weekly surveillance of automatic scram contactors. This is a significant safety improvement relative to the current operation of Quad Cities. The change to increase the frequency of testing the automatic scram contactor is a major safety improvement that is judged to be larger than any very small increase in risk associated with extending STIs or AOTs on other equipment. The licensee's proposed changes do not include changes to STIs and AOTs for the instruments of the reactor pressure vessel (RPV) steam dome high pressure and RPV low-low level functions because design changes to improve reliability of these instruments are expected to be completed during future refueling outages. Until the design changes are completed, the licensee has increased the calibration frequency for these instruments under the station administrative controls and has revised administrative procedures accordingly.

The licensee stated that a review of the recently upgraded Quad Cities probabilistic risk analysis (PRA) and its insights ensures that the proposed changes are acceptable. Also, because the proposed changes are limited to instrumentation STIs and AOTs, the increase in risk is very small. This is due to the high degree of redundancy and, in many cases, the diversity of instrumentation that provide automatic safety system actuation. Also, the industry PRA analyses, including those for Quad Cities, have generally confirmed that instrumentation for actuation of RPS, containment isolation, and ECCS is not a dominant contributor to risk. The licensee further added that Quad Cities has significant plant design features that reduce the impact of certain instrumentation failures. These features were not considered in the generic AOT and STI evaluations. If these features are considered in the plant-specific evaluations of the proposed changes, it will result in a further reduction of the already known low risk associated with STI and AOT extensions.

The site-specific evaluations (Refs. 10, 11, and 15) were performed by the licensee to support the proposed changes for safety systems and to assess the impact of the design differences between the specific plant and the generic base plant on safety when the proposed changes are implemented. Results of these site-specific evaluations indicated that the proposed changes would meet the acceptance criteria of the associated LTR, even though in few instances the safety systems configuration at Quad Cities, Units 1 and 2, differs from the configuration that was evaluated in the generic analyses of the base plant. In some cases, these differences will contribute to a small increase in failure frequency of the affected devices, but the effects of such a small increase in failure frequency were found to be within the acceptability guidelines of the associated LTR and would not significantly affect the improvement in plant safety achieved by implementing the proposed TS changes based on the generic plant analyses. The licensee stated that the generic evaluations were tested against the operating experience at Quad Cities to determine if it was appropriate to apply generic

changes to Quad Cities. If the evaluation determined a generic change to be inappropriate, the change was not applied to Quad Cities.

4.0 SUMMARY

On the basis of this evaluation, the staff concluded that the proposed changes will not alter assumptions relative to the mitigation of an accident or transient event, will not adversely affect normal plant operation and testing, will not alter the GE analysis conclusion that the proposed changes are consistent with the generic safety analyses and ISTSs, and that the overall effect of changes to the STIs and AOTs will be a net improvement in plant operation, without any significant increase in risk.

The staff noted that the licensee has met the conditions of the staff safety evaluation reports on LTRs by confirming that generic analysis results and conclusions are applicable to Quad Cities, the instrument drift characteristics are bounded by the assumptions used in the generic analysis for STI and AOT extensions, and the analysis for differences between parts of the RPS and the base plant has utilized the methodology outlined in the LTR for the RPS. Also, the upgraded PRA at the Quad Cities facility concluded that a change in risk as a result of implementation of the proposed changes is very insignificant. Therefore, the staff finds the proposed changes acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (65 FR 48746). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. NUREG-1433, Revision 1, "Standard Technical Specifications, General Electric Plants, BWR/4," April 1995.

2. Letter from C. L. Tully (BWROG) to B. Grimes (USNRC), BWROG-92102, dated November 4, 1992, "BWR Owners' Group (BWROG) Topical Reports on Technical Specification Improvement Analysis for BWR Reactor Protection Systems."

3. General Electric Licensing Topical Report NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.

4. General Electric Licensing Topical Report, NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

5. General Electric Licensing Topical Report, NEDC 31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

6. General Electric Licensing Topical Report, NEDC-30936P-A, Part 1 and Part 2, "Technical Specification Improvement Methodology With Demonstration for BWR ECCS Actuation Instrumentation," December 1988.

7. General Electric Licensing Topical Report, GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.

8. General Electric Licensing Topical Report, GENE 770-06-2-A, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.

9. General Electric Licensing Topical Report, NEDC-30851P-A, Supplement 1, "Technical Specification Improvement Analysis For BWR Control Rod Block Instrumentation," October 1988.

10. Technical Specification Improvement Analysis for the Reactor Protection System for Quad Cities Station, Units 1 and 2, Proprietary Reports No. GE-NE E11-00105-00-01-01 and GE-NE E11-00105-00-02-01, dated December 1999 (Proprietary is not publicly available); Non-proprietary Reports No. GE-NE E11-00105-00-05-01 and GE-NE E11-00105-00-06-01, dated March 2000.

11. Technical Specification Improvement Analysis For Isolation Actuation Instrumentation For Quad Cities Nuclear Power Station, Units 1 & 2, GE Report No. GE-NE E11-00105-00-04-01, dated December 1999.

12. Letter from C. Rossi (USNRC) to R. Janecek (BWROG) dated April 27, 1988, "Staff Guidance for Licensee Determination That Drift Characteristics for Instrumentation Used in RPS Channels are Bounded By NEDC-30851P Assumptions When the Functional Test Interval is Extended from Monthly to Quarterly."

13. BWROG letter to M. Wohl (USNRC) dated June 25, 1990, "Implementation Enhancements to Technical Specification Changes Given in Isolation Actuation Instrumentation Analysis."

14. W. Sullivan (GE) letter to M. Wohl (USNRC) dated March 22, 1990, "Clarification of Technical Specification Changes Given in ECCS Actuation Instrumentation Analysis."

15. Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for Quad Cities Nuclear Power Station, Units 1 and 2, Proprietary Report No. GE-NE E11-00105-00-03-01, dated December 1999 (Proprietary is not publicly available); Non-proprietary Report No. GE-NE E11-00105-00-07-01, dated March 2000.

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