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

August 2, 2000

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

SUBJECT: DRESDEN - ISSUANCE OF AMENDMENTS CHANGING ALLOWABLE OUT-OF-SERVICE TIMES AND SURVEILLANCE TEST INTERVALS (TAC NOS. MA7984 AND MA7985)

Dear Mr. Kingsley:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 177 to Facility Operating License No. DPR-19 and Amendment No. 173 to Facility Operating License No. DPR-25 for Dresden, Units 2 and 3. The amendments are in response to your application dated January 11, 2000.

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 have previously been reviewed and approved by NRC.

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

Sincerely,

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

Docket Nos. 50-237 and 50-249

Enclosures:
1. Amendment No. 177 to DPR-19
2. Amendment No. 173 to DPR-25
3. Safety Evaluation

cc w/encls: See next page

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DATE	07/20/00		08/1/00		07/ // /00		07/27/00	08/1/00

OFFICIAL RECORD COPY

cc:

Commonwealth Edison Company
Site Vice President - Dresden
6500 N. Dresden Road
Morris, Illinois 60450-9765

Commonwealth Edison Company
Dresden Station Manager
6500 N. Dresden Road
Morris, Illinois 60450-9765

U.S. Nuclear Regulatory Commission
Dresden Resident Inspectors Office
6500 N. Dresden Road
Morris, Illinois 60450-9766

Regional Administrator
U.S. NRC, Region III
801 Warrenville Road
Lisle, Illinois 60532-4351

Illinois Department of Nuclear Safety
Office of Nuclear Facility Safety
1035 Outer Park Drive
Springfield, Illinois 62704

Chairman
Grundy County Board
Administration Building
1320 Union Street
Morris, Illinois 60450

Document Control Desk-Licensing
Commonwealth Edison Company
1400 Opus Place, Suite 400
Downers Grove, Illinois 60515

Commonwealth Edison Company
Reg. Assurance Manager - Dresden
6500 N. Dresden Road
Morris, Illinois 60450-9765

Mr. David Helwig
Senior Vice President
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 900
Downers Grove, Illinois 60515

Mr. Gene H. Stanley
Vice President - Nuclear Operations
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 900
Downers Grove, Illinois 60515

Mr. Christopher Crane
Senior VP - Nuclear Operations
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 900
Downers Grove, Illinois 60515

Mr. R. M. Krich
Vice President - Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, Illinois 60515

Ms. Pamela B. Stroebel
Senior Vice President and General Counsel
Commonwealth Edison Company
P.O. Box 767
Chicago, Illinois 60690-0767

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 177
License No. DPR-19

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated January 11, 2000, 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 2.C.(2) of Facility Operating License No. DPR-19 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 177 , 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: August 2, 2000

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 173
License No. DPR-25

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated January 11, 2000, 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-25 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 173 , 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: August 2, 2000

ATTACHMENT TO LICENSE AMENDMENT NOS. 177 AND 173

FACILITY OPERATING LICENSE NOS. DPR-19 AND DPR-25

DOCKET NOS. 50-237 AND 50-249

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
3/4.1-1	3/4.1-1
- -	3/4.1-1a
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
B3/4.1-1	B3/4.1-1
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
- -	3/4.2-16b
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
3/4.2-26	3/4.2-26
3/4.2-32	3/4.2-32
3/4.2-33	3/4.2-33
3/4.2-34	3/4.2-34
3/4.2-48	3/4.2-48
3/4.2-49	3/4.2-49
3/4.2-51	3/4.2-51
B3/4.2-1	B3/4.2-1
B3/4.2-2	B3/4.2-2
B3/4.2-3	B3/4.2-3
B3/4.2-4	B3/4.2-4
B3/4.2-5	B3/4.2-5
- -	B3/4.2-6

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 CHANNEL(s) 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 CHANNEL(s) 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

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.

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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.

3.1 - LIMITING CONDITIONS FOR OPERATION

c. Within 12 hours, restore the inoperable CHANNEL(s) 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 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.

TABLE 3.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATION

(a) When a CHANNEL is placed in an inoperable status soley 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.

(j) This function is not required to be OPERABLE when reactor pressure is less than 600 psig.

TABLE 4.1.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	Functional Unit	Applicable OPERATIONAL MODES	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST[q]	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[l]	NA	M	Q
4.	Reactor Vessel Water Level - Low	1, 2	D	Q	E[h]
5.	Main Steam Line Isolation Valve - Closure	1, 2[p]	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[q]	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 (Unit 2), or Float Switch (Unit 3)	1, 2, 5[j,k]	NA	Q	NA
9.	Turbine Stop Valve - Closure	1[l]	NA	Q	E
10.	Turbine EHC Control Oil Pressure - Low	1[l]	NA	M	Q
11.	Turbine Control Valve Fast Closure	1[l]	NA	Q	E
12.	Turbine Condenser Vacuum - Low	1, 2[p]	NA	M	M
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 3/4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

(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) This function is not required to be OPERABLE when reactor pressure is less than 600 psig.

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

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 contactors 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.

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 CHANNEL(s) remain OPERABLE or in the tripped condition to ensure automatic isolation capability.

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.

3.2 - LIMITING CONDITIONS FOR OPERATION

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, 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 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 Function Unit shall be taken.

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) Deleted

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

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

(j) All four switches in either of 2 groups for each trip 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]	S	Q	E[a]	1, 2, 3 & *
b.	Drywell Pressure - High[b,c]	NA	Q	Q	1, 2, 3
c.	Reactor Building Ventilation Exhaust Radiation - High[c]	S	Q	Q	1, 2, 3 & * *
d.	Refueling Floor Radiation - High[c]	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	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.	ISOLATION CONDENSER				
a.	Steam Flow - High	NA	Q	Q	1, 2, 3
b.	Return Flow - High	NA	Q	Q	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.	SHUTDOWN COOLING ISOLATION				
a.	Reactor Vessel Water Level - Low	S	Q	E[(a)]	3, 4, 5
b.	Recirculation Line Water Temperature - High (Cut-in Permissive)	NA	Q	E	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) 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 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]	≤15' 5" above bottom of chamber	2	1, 2, 3	35
e.	Reactor Vessel Water Level - High (Trip)	≤194 inches	1	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 psig	2	1, 2, 3	38
c.	Initiation Timer	≤120 sec	1	1, 2, 3	31
d.	Low Low Level Timer	≤10 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 psig	2	1, 2, 3	38
c.	Initiation Timer	≤120 sec	1	1, 2, 3	31
d.	Low Low Level Timer	≤10 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
6.	LOSS OF POWER				
a.	4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	2930±146 volts decreasing voltage	2/bus	1, 2, 3, 4[e], 5[e]	36
b.	4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	≥ 3784 volts (Unit 2)[g][i] ≥ 3832 volts (Unit 3)[g][i]	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 IC or HPCI inoperable, restore the inoperable CHANNEL(s) to OPERABLE status within 96 hours or declare the ADS relief valves inoperable, AND

c. With IC 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.

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

ACTION

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 asociated 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.

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.

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

ACTION

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 IC 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	Q	E[e]	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	Q	E[e]	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	Q	E[e]	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	Q	E[e]	1, 2, 3
f.	HPCI Pump Discharge Flow - Low (Bypass)	NA	Q	Q	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	Q	E[e]	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 CHANNEL(s) per TRIP SYSTEM[a]
1.	Reactor Vessel Water Level - Low Low	$\geq$84 inches[b]	2
2.	Reactor Vessel Pressure - High	$\leq$1250 psig	2

---

a When a CHANNEL is placed in an inoperable status soley 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 3.2.D-1

ISOLATION CONDENSER ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per TRIP SYSTEM[a]	ACTION
Reactor Vessel Pressure - High	≤1070 psig	2	40

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 IC inoperable, AND

b. Place the inoperable CHANNEL(s) in the tripped condition within 24 hours or declare IC inoperable.

---

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 six hours provided the Function Unit maintains actuation capability.

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 $\geq$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 drive 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 $\times 10^6$ 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	Q	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)	$0.5 \leq p \leq 1.5$ psig	2	80
2. Reactor Vessel Water Level -Low (Permissive)	≥ -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 Sprays 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 Sprays 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 - High	NA	Q	Q
2. Reactor Vessel Water Level -Low	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.

TABLE 3.2.J-1

FEEDWATER PUMP TRIP INSTRUMENTATION

Functional Unit	Trip Setpoint[(a)]	Minimum CHANNEL(s)[(b)]	ACTION
Reactor Vessel Water Level -High	$\leq$ 201 inches	4	90

ACTION

ACTION 90 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:

a. Place the inoperable CHANNEL(s) in the tripped condition within 7 days AND

b. Within one hour from discovery of loss of feedwater pump trip capability, restore feedwater pump trip capability within 72 hours, or be in at least STARTUP within the next 8 hours.

---

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

b When a CHANNEL is placed in an operable 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 Feedwater Pump Trip capability.

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 active fuel and, thus the actual top of active fuel, when compared to the original fuel designs. Safety Limits, water level instrument 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 initiations 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).

BASES

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:

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 the 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 the 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 Electrical 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 the plant to this postulated event falls within the bounds of events studied in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO24222, dated December 1979. Tripping the recirculation pumps adds negative reactivity from the increase in 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 Isolation Condenser Actuation Instrumentation

The isolation condenser system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment. 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, Parts 1 and 2, "BWR Owner's Group Techical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation)," December 1998.

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 of 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

BASES

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 and STARTUP/HOT STANDBY OPERATIONAL MODE(s), the APRM rod block function setpoint is significantly reduced to provide the same type of protection in the REFUEL and STARTUP/HOT STANDBY OPERATIONAL 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.

BASES

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 the operators with the 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 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 (waste) holdup system to prevent a possible uncontrolled release via this pathway. This instrumentation is included in accordance with Generic Letter 89-01.

BASES

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 suppression chamber and drywell spray mode of the low pressure coolant injection/ containment cooling 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 that caused 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 177 TO FACILITY OPERATING LICENSE NO. DPR-19

AND AMENDMENT NO. 173 TO FACILITY OPERATING LICENSE NO. DPR-25

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated January 11, 2000, Commonwealth Edison Company (ComEd, the licensee) proposed license amendments to change the Technical Specifications (TSs) for Dresden Nuclear Power Station, Units 2 and 3 (Dresden). The proposed changes increase allowable out-of-service times (AOTs) and surveillance test intervals (STIs) for selected TS actuation instrumentation. The proposed changes implement recommendations resulting from generic evaluations of the licensing topical reports on AOTs and STIs performed by General Electric (GE) and the Boiling Water Reactor (BWR) Owners' Group and subsequently approved by the NRC. These tropical reports assessed the reliability of TS actuation instrumentation and concluded that extending AOTs and STIs for test and repair activities would enhance operational safety.

The proposed TS changes revise Sections 3/4.1, "Reactor Protection System," and 3/4.2, "Instrumentation," by increasing AOTs and STIs for specified actuation instrumentation. The proposed changes will permit channel functional tests to be conducted quarterly rather than weekly or monthly. In addition, the AOTs for repairs will be increased from 1 hour to 12 or 24 hours, and the AOTs for required surveillance tests will be increased from 2 hours to 6 hours. All proposed STI and AOT changes are consistent with GE licensing topical reports (LTRs) that have been reviewed and approved by the NRC. These topical reports assessed the reliability of TS actuation instrumentation and concluded that extending AOTs and STIs for test and repair activities would enhance operational safety.

2.0 BACKGROUND

In 1983 the Boiling Water Reactor Owners' Group (BWROG) formed a Technical Specifications Improvement (TSI) Committee. This committee established a program to identify improvements to allowable out-of-service times and surveillance test intervals specified in BWR Standard TS. The primary objective was to minimize unnecessary testing and restrictive out-of-service times that could potentially degrade overall plant safety and availability. Examples of

the problems with current TS are inadvertent scrams due to frequent testing, AOTs that are not of sufficient duration to perform repairs on a reasonable basis, excessive actuation of equipment contributing to component wear-out, and unwarranted radiation exposure to personnel performing surveillance testing.

During April 1984, the Committee met with the NRC and outlined the BWR TS Improvement Program. The NRC expressed agreement with the overall approach. Subsequently, the BWROG developed a series of LTRs to provide the basis for extending the AOTs and STIs for key actuation instrumentation, including the reactor protection system (RPS), the emergency core cooling system (ECCS), containment isolations, control rod block functions, and other miscellaneous functions. These GE LTRs were reviewed and approved by the NRC. The NRC required that plant-specific applications confirm the applicability of the generic analyses to the specific plant, and confirm that setpoint drift, which could be expected under the extended test intervals, would remain within the existing allowances in the respective instrument setpoint calculations.

3.0 EVALUATION

3.1 Justification for Changes

The proposed increase in STIs for surveillance and AOTs for repair reflect Standard Technical Specification (STS) revisions recommended in the LTRs and are based upon reliability analyses. These changes are beneficial in reducing (1) potential unnecessary plant scrams, (2) excessive equipment test cycles, and (3) the diversion of personnel and resources for unnecessary testing. The NRC staff has reviewed and approved these LTRs in safety evaluation reports (SERs). Subsequently, all the LTRs were issued with the corresponding SER included.

3.1.1 Plant-Specific LTRs

GE Report NEDC-30851P-A, March 1988, provides an acceptable generic basis for supporting plant-specific TS changes related to the extensions of STIs and AOTs for the RPS, isolation actuation, and the ECCS instrumentation.

(1) GE-NE E11-00084-1-01, "Technical Specification Improvements for the Reactor Protection System Instrumentation of the Dresden Nuclear Power Station, Units 2 & 3," December 1999.

This report extends the generic study of modifying the TS requirements of RPS on a plant specific basis for Dresden. The generic study provides a technical basis to modify the surveillance test frequencies and AOTs of the RPS from the generic TSs. The generic study also provides additional analyses of various known different RPS configurations to support the application of the generic basis on a plant specific basis. The generic basis and the supporting analyses were utilized in this plant specific evaluation. The plant specific evaluation of the surveillance test frequencies and AOT of the RPS was performed as required in the acceptable generic basis. The results of the plant specific evaluation of the RPS for Dresden are identified in this report. The results

indicated that the RPS configuration for Dresden has several differences to the RPS configuration in the generic evaluation. The NRC has evaluated these differences and the assessment of their effects on the RPS failure frequency and concludes that these differences would not significantly affect the improvement in plant safety due to the changes in the TSs based on the generic analysis. Therefore, the generic analysis is applicable to Dresden.

(2) GE-NE E11-00084-04, "Technical Specification Improvements for the Isolation Actuation Instrumentation of the Dresden Nuclear Power Station, Units 2 & 3," March 1997.

This report provides an analysis of the isolation actuation instrumentation common to the RPS and ECCS instrumentation. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990, and NEDC-30851P-A Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989, provided an acceptable basis for extending surveillance test intervals and allowable outage times for isolation actuation instrumentation. Bases are provided for extending the isolation actuation instrumentation surveillance test intervals from 1 to 3 months and AOTs for tests and repairs from 2 hours and 1 hour to 6 hours and 24 hours, respectively. These generic studies provide bounding analyses of the impact of proposed TS changes for isolation actuation instrumentation and also provide verification that the results of the generic analyses of the various product lines are applicable to the individual plant TS requirements. This evaluation included a comparison of isolation actuation instrumentation STIs and calibration intervals given in the current plant-specific TS to those evaluated for the product lines. The licensee evaluated the differences to verify that the product line analyses envelope these differences.

The majority of the plants in the generic study have the same logic and number of sensor variables that initiate system isolation. The licensee evaluated the differences in the number of sensor variables and justified that the effect of these differences on the overall analysis results is insignificant. The NRC agrees to the results from case studies performed for different types of instrumentation logic and number of sensor variables. Therefore, the generic analyses of NEDC-30851P-A and NEDC-31677P-A are applicable to Dresden and provide an adequate basis for TS changes to extend the STIs and AOTs for Dresden isolation actuation instrumentation.

(3) GE-NE E11-00084-02, "Technical Specification Improvements for the Emergency Core Cooling System Instrumentation of the Dresden Nuclear Power Station, Units 2 & 3," December 1999.

This report extends the generic study of modifying the TS requirements of the ECCS on a plant specific basis for Dresden. The generic study provides a technical basis to modify the STIs and AOTs of the ECCS actuation instrumentation from those of the generic TS. The generic study also provides additional analyses of various known different ECCS configurations to support the application of the generic basis on a plant specific basis.

A plant specific evaluation of modifying the STIs and AOTs of the ECCS from the TSs of Dresden, followed the procedures of NEDC-30936P-A and has been performed. The result indicates that the ECCS configuration for Dresden is similar to the ECCS configuration in the generic evaluation, with 11 differences. These differences have been evaluated to show that the proposed changes to ECCS actuation instrumentation TSs would meet the acceptance criteria in NEDC-30936P-A. The results of our review indicate that the differences and their impact do not affect the applicability of the TS changes developed by the generic basis. Therefore, the generic analysis in NEDC-30936P-A is applicable to Dresden.

3.1.2 Generic LTRs

ComEd has reviewed and determined that the generic analyses performed by GE for the BWROG to revise AOTs and STIs are applicable to Dresden. The licensee has completed the necessary plant-specific evaluations for Dresden required by the NRC SERs. The following discussion provides the information required by the NRC for plant-specific submittals. As stated in the SERs for the AOT/STI LTRs, three conditions for the RPS LTR (3.2, (3)) identified in Items 1, 2 and 3 below, and two conditions for the other LTRs (3.2, (4) through (8)) identified in Items 1 and 2 below, must be addressed to justify application of the generic analyses to an individual plant when specific TSs are considered for revision. These issues are addressed as follows:

(1) Confirm the applicability of the generic analyses to the plant.

(a) The general study in Licensing Topical Report NEDC-30851P-A (3.2, (3)) provides a technical basis to modify the surveillance test frequencies and AOT of the RPS. This LTR identifies ComEd (including Dresden) as a participating utility in the development of the RPS TS Improvement Analysis. The generic study also provides additional analyses of different RPS configurations to support the application of the generic basis to specific plants. A plant-specific evaluation for modifying the surveillance test frequencies and AOT of the RPS in the Dresden TS has been performed and is contained in the plant-specific evaluation report (3.1, (9)). The evaluation utilized the generic basis and the additional analyses documented in LTR NEDC-30851P-A. In order to comply with the results of the NEDO-30851P-A analysis for RPS, Dresden has included in the proposed TS change the requirement to test, on a weekly basis, the automatic scram contactors. This testing will be accomplished using the RPS system subchannel keylock switches, which de-energize the associated automatic scram contactor relays.

(b) Licensing Topical Report NEDC-30851P-A, Supplement 2 (3.2, (4)), provides the justification for TS improvements for BWR isolation functions common to the RPS and ECCS. This LTR identifies ComEd, including Dresden, as a participating utility in the development of the BWR Isolation Instrumentation common to the RPS and ECCS Technical Specification Improvement Analysis. Also, this report specifically addresses BWR 3/4 plants and includes common isolation functions consistent with those at Dresden. Furthermore, this report

includes a plant-specific evaluation performed by GE for isolation actuation functions (3.2, (10)). The evaluation confirms the applicability of the generic analysis to Dresden.

(c) Licensing Topical Report NEDC-31677P-A (3.2, (5)) provides the justification for TS improvements for BWR isolation functions not common to the RPS or ECCS. This report identifies ComEd (including Dresden) as a participating utility in the development of the analysis in this report. The results for the BWR/3 product line are presented in this report. Further, this report includes a plant-specific evaluation performed by GE for isolation actuation functions. The evaluation confirms the applicability of the generic analysis to Dresden.

(d) Licensing Topical Report NEDC-30936P-A, Parts 1 and 2 (3.2, (6)), provide the justification for TS improvements for BWR ECCS functions. ComEd participates in the development of the BWROG Technical Specification Improvement Methodology for ECCS Actuation Instrumentation. This report describes the generic analyses performed for BWR 3/4 plants. In order to support the full BWR product line, GE identified major differences from the population of BWRs that participated in the AOT/STI program. The differences were evaluated by changing the generic fault trees and reevaluating the impact of proposed AOT/STI changes. These are documented as case studies in Topical Report NEDC-30936P-A. Using the results of the case studies, individual plant-specific reports were produced to support each individual utility's submittal to the NRC.

For Dresden site-specific analysis, GE compared the Dresden design configuration with the configuration of the case study plant. Only a single difference required additional analysis; the effect of the 18-month surveillance interval for the ADS drywell pressure bypass timers and ADS initiation timers. The case study assumed a 1-month surveillance interval. GE determined the effect of this change to be insignificant. Included in this report is the plant-specific evaluation provided by GE for ECCS actuation functions (3.1, (14)). The evaluation confirms the applicability of the general analysis to Dresden.

(e) Licensing Topical Report GENE-770-06-1-A (3.2, (7)) provides the justification for TS improvements for selected instrumentation functions. This report identifies changes to STIs and AOTs for selected instrumentation for BWR plants. The report concluded that extending functional test frequencies and AOTs for certain instruments was appropriate. The site-specific actuation functions encompassed by the report are ATWS-RPT and suppression chamber and drywell spray actuation. GE has performed a plant-specific evaluation that confirms the applicability of the report analysis to Dresden as summarized below:

ATWS-RPT: Instrumentation for the ATWS-RPT actuation consists of two low-low RPV level instruments and two high RPV pressure instruments. The signals are combined in a two-out-of-two taken once logic scheme. The report analysis evaluated the following logic schemes: one-out-of-two taken twice and two-out-of-two taken once. The design of Dresden is consistent with the logic schemes

of the LTR. The report evaluation concluded that the proposed changes to ATWS instrumentation AOTs and STIs have a negligible effect on the reactivity shutdown failure frequency. For these reasons, the proposed changes are acceptable.

Suppression Chamber and Drywell Spray Actuation: As stated in this report, certain BWRs employ manually initiated suppression pool and drywell spray functions. This design approach is consistent with the Dresden containment spray functions. The evaluation in this report concluded that extending the AOTs and STIs for these functions is acceptable because there are no automatic functions and, therefore, no corresponding automatic initiation functions. Similar extensions of AOTs and STIs for automatic actuation functions were found to have a negligible impact on plant safety.

(f) Licensing Topical Report NEDC-30851P-A, Supplement 1 (3.2, (8)), provides the TS improvement analysis for control rod block actuation functions. The licensee participated in the development of the TS improvement analysis of BWR control rod block instrumentation for this report. Extending the AOTs and STIs for the control rod block actuation functions was found acceptable due to the benefits associated with reduced testing. LTR GENE-770-06-1-A (3.2, (7)) provides the basis for extending specific control rod block AOTs for testing and maintenance activities. As with RPS, extending the AOTs and STIs for the control rod block functions has no significant impact on the availability of the control rod block functions. Therefore, the proposed changes have a negligible impact on reactor safety. GE has performed a plant-specific evaluation that confirms the applicability of this report analysis to Dresden.

(2) Demonstrate, by use of current drift information provided by the equipment vendor or plant specific data, that the drift characteristics for instrumentation used in the channels in the plant are bounded by the assumptions used in the generic analyses when the functional test interval is extended from monthly (or weekly) to quarterly.

The LTRs for the AOT/STI do not contain specific instrument drift assumptions. For this reason, the requirements for plant-specific applications were clarified in a letter from C.E. Rossi (NRC) to R.F. Janecek (BWROG) dated April 27, 1988. In the letter, the NRC requested licensees to confirm that the setpoint drift which could be expected under the extended STIs has been studied and either (1) has been shown to remain within the existing allowance in the RPS and ESFAS instrument setpoint calculation, or (2) that the allowance and setpoint have been adjusted to account for the additional expected drift.

The setpoint methodology at Dresden uses the instrument calibration frequency to account for potential instrument drift. For the instruments within the scope of AOT/STI, the calibration intervals are not being modified; therefore, the proposed changes are within the existing allowances in the instrument setpoint calculations. The only exceptions are instrument loops that contain Rosemount analog trip units. These devices are calibrated every 31 days in accordance with the current TS. The licensee

has explicitly evaluated the setpoint drift associated with the Rosemount analog trip units. The results confirm that extending the current calibration frequency from 31 days to 92 days is acceptable and within existing setpoint allowances. This is consistent with the evaluation of analog trip units provided in 3.2, (3), which states: "Current vendor drift information on analog trip units indicates that the calibration interval could be extended to 6 months." Therefore, increasing the channel functional test intervals in accordance with the AOT/STI LTR and, where applicable, the Rosemount analog trip unit calibration intervals from 31 days to 92 days is consistent with the current setpoint methodology at Dresden. Based on our review, we find that the licensee has adequately addressed drift problem associated with increasing STIs from monthly to quarterly for LTRs (3.2, (3) through 3.2, (8)).

(3) Confirm that the differences between the parts of the RPS that perform the trip functions in the plant and those of the base case plant included in the analysis for its plant, were done using the procedures of Appendix K of NEDC-30851P (3.2, (3)).

The analysis utilized the methodology outlined in Appendix K of NEDC-30851P to identify and evaluate the differences between the parts of the RPS that perform the trip functions at Dresden and those analyzed in the generic study. The results of the site-specific analysis indicate that although the RPS configurations for Dresden differ in some respects from the configuration of the base case plant, the differences do not have a significant impact on the results and conclusions of the generic evaluation. Therefore, the generic analysis is applicable to Dresden. Furthermore, as required by NEDC-30851P, the proposed TS changes include a weekly surveillance of the automatic scram contactors.

3.2 Acceptance Basis

The Dresden TS requirements for STIs and AOTs are to be revised for the RPS, ECCS actuation, isolation actuation, anticipated transient without scram (ATWS) - recirculation pump trip (RPT), isolation condenser (IC) actuation instrumentation, control rod block actuation, suppression chamber and drywell spray actuation, and feedwater pump trip instrumentation. The revisions are based on the following documents:

(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 (NRC), BWROG-92102, "BWR Owners' Group (BWROG) Topical Reports on Technical Specification Improvement Analysis for BWR Reactor Protection Systems," dated November 4, 1992.

(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, NEDC-30851-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

(9) General Electric Report, "Technical Specification Improvements for the Reactor Protection System for the Dresden Nuclear Power Station, Units 2 and 3;" Proprietary Report No. GE-NE E11-00084-01-01, December 1999; Non-proprietary Report No. GE-NE E11-00084-01-03, March 2000.

(10) General Electric Report No. GE-NE E11-0084-04, "Technical Specification Improvement Analysis for Isolation Actuation Instrumentation for Dresden Nuclear Power Station, Units 2 and 3," March 1997.

(11) Letter from C. Rossi (NRC) 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."

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

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

(14) General Electric Report, "Technical Specification Improvements for the Emergency Core Cooling System Instrumentation of the Dresden Nuclear Power Station, Units 2 and 3;" Proprietary Report No. GE-NE E11-00084-01-02, December 1999; Non-proprietary Report No. GE-NE E11-00084-01-04, March 2000.

In NUREG-1024, "Technical Specifications Enhancing the Safety Impact," the NRC suggested that TS action statements be reviewed to assure that they have an adequate technical basis and do indeed minimized plant risk. The use of reliability analyses to support engineering judgment was recognized as a primary basis for improving TS requirements. Consistent with this approach, BWROG generated a series of LTRs justifying AOT and STI extensions in the TS for the RPS, isolation system, ECCS, and control rod block system instrumentation.

3.3 Evaluation of Technical Specification Changes

The following TS change descriptions are grouped by TS Section:

(1) Page 3/4.1-1: TS 3.1, A. Reactor Protection System, Actions 1 and 2 and Footnotes a and b are revised to incorporate a 1-hour check for trip capability and to allow a 12-hour AOT for maintenance activities. Based on 3.2, (2) and (3), these proposed changes are acceptable.

(2) Page 3/4.1-6: TS Table 3.1.A-1, Note (a) is revised to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability. Based on 3.2, (3), this proposed change is acceptable and is also consistent with the improved Standard Technical Specification (iSTS) (3.2, (1)).

(3) Page 3/4.1-7: Note (q) is added to TS Table 4.1.A-1 to provide for weekly testing of automatic scram contactors. As discussed in 3.2, (3), sensor channel functional tests can be done quarterly provided the automatic scram contactors are tested weekly. Based on 3.2, (3), this proposed change is acceptable.

(4) Pages 3/4.1-7, 3/4.1-8, and 3/4.1-9: TS Table 4.1.A-1 is revised to extend the channel functional test frequency to quarterly for the Functional Units 2.b, 2.c, 2.d, 4, 5, 7, 9, 11, and 12. The channel calibration frequency for Functional Unit 12 has been revised to monthly to reflect the current station practices. TS Table 4.1.A-1, Note (h), has been changed to reflect a 92-day calibration of associated trip units. Based on 3.2 (3), these proposed changes are acceptable.

(5) Pages 3/4.2-1 and 3/4.2-2: Action 2 is revised 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 the RPS and AOTs of 24 hours to repair functional units that are not common to the RPS. Footnote (a) is revised to reflect a change in the AOT requirement. Because of 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. Based on 3.2, (4), (5), (10), and (12), these proposed changes are acceptable.

(6) Page 3/4.2-7: TS Table 3.2.A-1, Note (a) is revised to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability. Based on 3.2, (4), (5), and (10), these proposed changes are acceptable and are also consistent with the iSTS (3.2, (1)).

(7) Pages 3/4.2-8, 3/4.2-9, and 3/4.2-10: TS Table 4.2.A-1 is revised 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.c, 3.d, 4.b, 5.a, 6.a, 6.b, 7.a, and 7.b. TS 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. Based on 3.2, (4), (5), and (10), these proposed changes are acceptable.

(8) Pages 3/4.2-11 and 3/4.2-12: Action 3 is deleted. The Action 3 requirements have been incorporated into a new Action 38 for automatic depressurization system (ADS) initiation instrumentation. This change is editorial in nature and is also consistent with the iSTS (3.2, (1)), and, therefore, is acceptable.

(9) Pages 3/4.2-14 and 3/4.2-15: For ADS permissive functions 4e, 4f, and 5f, the Minimum Channels per Trip Function requirement has been increased from 1/pump to 2/pump. This change is restrictive to ensure that each ADS trip system has a sufficient number of operable channels to initiate during a design basis event. Therefore, this change is acceptable and is also consistent with the iSTS (3.2, (1)).

(10) Page 3/4.2-14: A new Action 37 is proposed for HPCI Initiation Functional Units 3a and 3b. This change is conservative to ensure HPCI injection capability and is also consistent with the iSTS (3.2, (1)), and, therefore, is acceptable.

(11) Pages 3/4.2-14 and 3/4.2-15: A new Action 38 is proposed to incorporate ADS initiation functions 4a, 4b, 5a, and 5b. This Action is developed using LTR (3.2, (6)) and is also consistent with iSTS (3.2, (1)) and, therefore, is acceptable.

(12) Page 3/4.2-16: Action 30 is revised 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. Based on 3.2, (6), (13), and (14), this change is acceptable and is also consistent with the iSTS (3.2, (1)).

(13) Page 3/4.2-16: Action 31 is revised 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. Based on 3.2, (6), (13), and (14), this change is acceptable and is also consistent with the iSTS (3.2, (1)).

(14) Page 3/4.2-16: Action 32 is revised to extend the AOT for maintenance activities to 24 hours. Based on 3.2, (6), (13), and (14), this change is acceptable.

(15) Page 3/4.2-16: Action 33 is revised 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 low. This change is editorial in nature and is also consistent with the iSTS (3.2, (1)), and, therefore, is acceptable.

(16) Page 3/4.2-16: Action 34 is revised to extend the AOT for maintenance activities to 24 hours. Based on 3.2, (6), (13), and (14), this change is acceptable.

(17) Page 3/4.2-16: Action 35 is revised to incorporate a 1-hour loss-of-function check and a 24-hour AOT for maintenance activities. Based on 3.2, (6), (13), and (14), this change is acceptable, and is also consistent with the iSTS (3.2, (1)).

(18) Pages 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. Based on 3.2, (6), (13), and (14), this change is acceptable, and is also consistent with the iSTS (3.2, (1)).

(19) Page 3/4.2-16: A new Action 38 is proposed for ADS initiation functions 4a, 4b, 5a, and 5b. This proposed 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). Based on 3.2, (6), (13), and (14), this change is acceptable and is also consistent with the iSTS (3.2, (1)).

(20) Page 3/4.2-17: TS Table 3.2.B-1, Note (a) is revised to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability. Based on 3.2, (6) and (14), these changes are acceptable and are also consistent with the iSTS (3.2, (1)).

(21) Pages 3/4.2-18, 3/4.2-19, and 3/4.1-20: TS Table 4.2B-1 is revised 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, 3.a, 3.b, 3.c, 3.d, 3.e, 4.a, 4.b, 4.e, and 4.f. Based on 3.2, (6) and (14), this change is acceptable.

(22) Page 3/4.2-23: TS Table 3.2.C-1, Note (a) is revised to allow a 6-hour AOT for testing and to incorporate a check for trip capability. Based on 3.2, (7), this change is acceptable and is also consistent with the iSTS (3.2, (1)).

(23) Page 3/4.2-26: TS Table 3.2.D-1, Note (a) is revised to allow a 6-hour AOT for testing and to incorporate a check for trip capability. Based on 3.2, (6), these changes are acceptable, and are also consistent with the iSTS (3.2, (1)) and the reliability analysis approved by the NRC.

(24) Page 3/4.2-26: Action 40 related to the Isolation Condenser is revised to incorporate a 1-hour loss-of-function check and a 24-hour AOT for maintenance activities. Based on 3.2, (6), this change is acceptable, and is also consistent with the iSTS (3.2, (1)).

(25) Page 3/4.2-32: TS Table 3.2.E-1, Action 52 is revised to increase the AOT for maintenance activities from 1 hour to 12 hours. Based on 3.2, (7), this change is acceptable.

(26) Page 3/4.2-33: TS Table E-1, Note (i) is revised to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability. Based on 3.2, (7), this change is acceptable and is also consistent with the iSTS (3.2, (1)).

(27) Page 3/4.2-34: TS Table 4.2.E-1 is revised 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. Based on 3.2, (8), this change is acceptable.

(28) Page 3/4.2-48: TS Table 3.2.1-1, Action 80.a, is revised to increase the AOT for maintenance activities from 1 hour to 24 hours. Based on 3.2, (7), this change is acceptable.

(29) Page 3/4.2-48: TS Table 3.2.1-1, Note (c) is revised to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability. Based on 3.2, (7), this change is acceptable and is also consistent with the iSTS (3.2, (1)).

(30) Page 3/4.2-49: TS Table 4.2.1-1 is revised to increase the channel functional test interval from M to Q for Functional Units 1 and 2. In addition, Table 4.2.1-1, Note (a) has been changed to reflect a 92-day calibration of the associated trip units. Based on 3.2, (7), this change is acceptable.

(31) Page 3/4.2-51: TS Table 3.2.J-1, Feedwater Pump Trip Instrumentation - the minimum channel column is revised to require four instruments instead of two to reflect a previously installed design upgrade, whereby the Yarways were replaced with analog transmitter and trip units. Action 90 is revised to incorporate a 2-hour loss-of-function check but maintains the 7-day AOT for an inoperable instrument. Table 3.2.J-1, Note (b) is revised to increase the AOT for required surveillance testing from 2 hours to 6 hours and to incorporate a check for trip capability. Based on 3.2, (7), these changes are acceptable and are also consistent with the iSTS (3.2, (1)).

(32) The licensee has revised the Bases Sections of TS to be consistent with the changes identified in the STIs and AOTs.

4.0 SUMMARY

The proposed changes extend STIs and AOTs for instrumentation and have been justified using probabilistic analytical methods. The affected instrumentation is associated with the RPS, the ECCS, containment isolations, control rod block functions, and other miscellaneous functions. The changes have been the subject of generic licensing topical reports which the NRC has reviewed and approved. Dresden Nuclear Power Station has addressed the implementation of the generic TS changes identified in the LTRs on a plant-specific basis. The staff has reviewed the LTRs and the plant-specific reports and concludes that the generic analyses are applicable to Dresden. Also, the licensee performed the required plant-specific analysis and justified the application of the generic analysis to Dresden. The information on setpoint drift supports the conclusion that setpoint drift is within the existing allowances in the respective instrument setpoint calculations under the extended test intervals. Therefore, we have found the proposed changes to the Dresden TS and Bases 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 and change surveillance requirements. 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 12290). 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.

Principal Contributor: Sang Rhow, EEIB

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