



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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 95  
License No. NPF-11

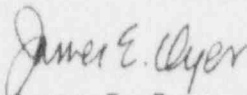
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated January 28, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 set forth in 10 CFR Chapter I;
  - 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 enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-11 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 95, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective upon date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



James E. Dyer, Director  
Project Directorate III-2  
Division of Reactor Projects - III/IV/V  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: February 25, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 95

FACILITY OPERATING LICENSE NO. NPF-11

DOCKET NO. 50-373

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain a vertical line indicating the area of change.

REMOVE

3/4 3-7  
3/4 3-8  
3/4 3-44  
B 3/4 3-1  
B 3/4 3-3  
-

INSERT

3/4 3-7  
3/4 3-8  
3/4 3-44  
B 3/4 3-1  
B 3/4 3-3  
B 3/4 3-3a

TABLE 4.3.1.1-1

## REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION <sup>(a)</sup>	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors				
a. Neutron Flux - High	S/U <sup>(b)</sup> , S S	S/U <sup>(c)</sup> , W W	R R	2 3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor: <sup>(f)</sup>				
a. Neutron Flux - High, Setdown	S/U <sup>(b)</sup> , S S	S/U <sup>(c)</sup> , W W	SA SA	1, 2 3, 5
b. Flow Biased Simulated Thermal Power-Upscale	S, D <sup>(g)</sup>	S/U <sup>(c)</sup> , W	W <sup>(d)(e)</sup> , SA, R <sup>(h)</sup>	1
c. Fixed Neutron Flux - High	S	S/U <sup>(c)</sup> , W	W <sup>(d)</sup> , SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	NA	M	Q	1, 2
4. Reactor Vessel Water Level - Low, Level 3	NA	M	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
6. Main Steam Line Radiation - High	S	M	R	1, 2
7. Primary Containment Pressure - High	NA	M	Q	1, 2

TABLE 4.3.1.1-1 (Continued)

## REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	NA	M	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	NA	1, 2, 3, 4, 5
13. Control Rod Drive				
a. Charging Water Header Pressure - Low	NA	M	R	2, 5
b. Delay Timer	NA	M	R	2, 5

- (a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
- (b) The IRM, and SRM channels shall be determined to overlap for at least 1/2 decades during each startup and the IRM and APRM channels shall be determined to overlap for at least 1/2 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.
- (d) This calibration shall consist of the adjustment of the APRM channel to conform to the power levels calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER  $\geq$  25% of RATED THERMAL POWER. The APRM Gain Adjustment Factor (GAF) for any channel shall be equal to the power value determined by the heat balance divided by the APRM reading for that channel.

Within 2 hours, adjust any APRM channel with a GAF  $>$  1.02. In addition, adjust any APRM channel within 12 hours, (1) if power is greater than or equal to 90% of RATED THERMAL POWER and the APRM channel GAF is  $<$  0.98, or (2) if power is less than 90% of RATED THERMAL POWER and the APRM reading exceeds the power value determined by the heat balance by more than 10% of RATED THERMAL POWER. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

- (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 1000 effective full power hours (EFPH).
- (g) Measure and compare core flow to rated core flow.
- (h) This calibration shall consist of verifying the  $6 \pm 1$  second simulated thermal power time constant.

TABLE 4.3.4.2.J-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL CALIBRATION</u>
1. Turbine Stop Valve-Closure	Q	R
2. Turbine Control Valve-Fast Closure	Q	R

### 3/4.3 INSTRUMENTATION

#### BASES

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#### 3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

- a. Preserve the integrity of the fuel cladding.
- b. Preserve the integrity of the reactor coolant system.
- c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and
- d. Prevent inadvertent criticality.

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 channels 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 systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279, 1971, for nuclear power plant protection systems. Specified surveillance intervals for MSIV-Closure, TSV-Closure, TCV-Closure, and the Manual Scram have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System", March 1988. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the accident analysis. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) in-place, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

## INSTRUMENTATION

### BASES

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#### 3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December, 1979, and Appendix G of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A generic analysis, which provides for continued operation with one or both trip systems of the EOC-RPT system inoperable, has been performed. The analysis determined bounding cycle independent MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) values which must be used if the EOC-RPT system is inoperable. These values ensure that adequate reactivity margin to the MCPR safety limit exists in the event of the analyzed transient with the RPT function inoperable. The analysis results are further discussed in the bases for Specification 3.2.3.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

Specified surveillance intervals have been determined in accordance with the following:

1. NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System", March 1988.



## INSTRUMENTATION

### BASES

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#### 3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

2. 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.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 190 ms, less the time allotted for sensor response, i.e., 10 ms, and less the time allotted for breaker arc suppression determined by test, as correlated to manufacturer's test results, i.e., 83 ms, and plant pre-operational test results.



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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 79  
License No. NPF-18

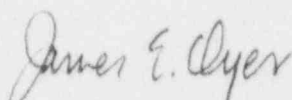
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated January 28, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 set forth in 10 CFR Chapter I;
  - 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 enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-18 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 79 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective upon date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



James E. Dyer, Director  
Project Directorate III-2  
Division of Reactor Projects - III/IV/V  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: February 25, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 79

FACILITY OPERATING LICENSE NO. NPF-18

DOCKET NO. 50-374

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain a vertical line indicating the area of change.

<u>REMOVE</u>	<u>INSERT</u>
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
3/4 3-44	3/4 3-44
B 3/4 3-1	B 3/4 3-1
B 3/4 3-3	B 3/4 3-3
-	B 3/4 3-3a

TABLE 4.3.1.1-1

## REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL CALIBRATION(a)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>
1. Intermediate Range Monitors				
a. Neutron Flux - High	S/U <sup>(b)</sup> , S	S/U <sup>(c)</sup> , W	R	2
	S	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor: <sup>(f)</sup>				
a. Neutron Flux - High, Setdown	S/U <sup>(b)</sup> , S	S/U <sup>(c)</sup> , W	SA	1, 2
	S	W	SA	3, 5
b. Flow Biased Simulated Thermal Power-Upscale	S, D <sup>(g)</sup>	S/U <sup>(c)</sup> , W	W <sup>(d)(e)</sup> , SA, R <sup>(h)</sup>	1
c. Fixed Neutron Flux - High	S	S/U <sup>(c)</sup> , W	W <sup>(d)</sup> , SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	NA	M	Q	1, 2
4. Reactor Vessel Water Level - Low, Level 3	NA	M	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
6. Main Steam Line Radiation - High	S	M	R	1, 2
7. Primary Containment Pressure - High	NA	M	Q	1, 2

TABLE 4.3.1.1-1 (Continued)

## REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	NA	M	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	NA	1, 2, 3, 4, 5
13. Control Rod Drive				
a. Charging Water Header Pressure - Low	NA	M	R	2, 5
b. Delay Timer	NA	M	R	2, 5

- (a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
- (b) The IRM, and SRM channels shall be determined to overlap for at least 1/2 decades during each startup and the IRM and APRM channels shall be determined to overlap for at least 1/2 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.
- (d) This calibration shall consist of the adjustment of the APRM channel to conform to the power levels calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER  $\geq$  25% of RATED THERMAL POWER. The APRM Gain Adjustment Factor (GAF) for any channel shall be equal to the power value determined by the heat balance divided by the APRM reading for that channel.

Within 2 hours, adjust any APRM channel with a GAF  $>$  1.02. In addition, adjust any APRM channel within 12 hours, (1) if power is greater than or equal to 90% of RATED THERMAL POWER and the APRM channel GAF is  $<$  0.98, or (2) if power is less than 90% of RATED THERMAL POWER and the APRM reading exceeds the power value determined by the heat balance by more than 10% of RATED THERMAL POWER. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

- (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 1000 effective full power hours (EFPH).
- (g) Measure and compare core flow to rated core flow.
- (h) This calibration shall consist of verifying the  $6 \pm 1$  second simulated thermal power time constant.

TABLE 4.3.4.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL CALIBRATION</u>
1. Turbine Stop Valve Closure	Q	R
2. Turbine Control Valve-Fast Closure	Q	R

### 3/4.3 INSTRUMENTATION

#### BASES

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#### 3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

- a. Preserve the integrity of the fuel cladding.
- b. Preserve the integrity of the reactor coolant system.
- c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and
- d. Prevent inadvertent criticality.

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 channels 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 systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279, 1971, for nuclear power plant protection systems. Specified surveillance intervals for MSIV-Closure, TSV-Closure, TCV-Closure, and the Manual Scram have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System", March 1988. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the accident analysis. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) in place, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.



## INSTRUMENTATION

### BASES

#### 3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December, 1979, and Appendix G of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A generic analysis, which provides for continued operation with one or both trip systems of the EOC-RPT system inoperable, has been performed. The analysis determined bounding cycle independent MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) values which must be used if the EOC-RPT system is inoperable. These values ensure that adequate reactivity margin to the MCPR safety limit exists in the event of the analyzed transient with the RPT function inoperable. The analysis results are further discussed in the bases for Specification 3.2.3.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

Specified surveillance intervals have been determined in accordance with the following:

1. NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System", March 1988.

## INSTRUMENTATION

### BASES

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#### 3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

2. 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.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 190 ms, less the time allotted for sensor response, i.e., 10 ms, and less the time allotted for breaker arc suppression determined by test, as correlated to manufacturer's test results, i.e., 83 ms, and plant pre-operational test results.