50-454/50-455
50-456/50-457

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

June 26, 1996

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M90529, M90530, M90531 AND M90532)

Dear Mr. Farrar:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 84 to Facility Operating License No. NPF-37 and Amendment No. 84 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 76 to Facility Operating License No. NPF-72 and Amendment No. 76 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated September 16, 1994, as supplemented by letter dated January 31, 1996.

The amendments revise the technical specifications to eliminate periodic response time testing requirements for selected pressure and differential pressure sensors in the reactor trip system and engineered safety features actuation instrumentation channels.

In the January 31, 1996 submittal, ComEd committed to take certain actions when eliminating pressure and differential pressure sensor response time testing requirements. The actions are consistent with those in the staff's conditional approval of WCAP-13632-P-A, "Elimination of Pressure Sensor Response Time Testing Requirements," Revision 2. Please notify us when those actions are completed.

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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,

/s/

George F. Dick, Jr., Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-454, STN 50-455,
STN 50-456 and STN 50-457

Enclosures:
1. Amendment No. 84 to NPF-37
2. Amendment No. 84 to NPF-66
3. Amendment No. 76 to NPF-72
4. Amendment No. 76 to NPF-77
5. Safety Evaluation

cc w/encl: see next page

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 84
License No. NPF-37

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 16, 1994, as supplemented by letter dated January 31, 1996, 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. NPF-37 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 84 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and is to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George F Dick

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 26, 1996

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 84
License No. NPF-66

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 16, 1994, as supplemented by letter dated January 31, 1996, 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 1;

   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. NPF-66 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 84 and revised by Attachment 2 to NPF-66, and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. Attachment 2 contains a revision to Appendix A which is hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and is to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 26, 1996

ATTACHMENT TO LICENSE AMENDMENT NOS. 84 AND 84

FACILITY OPERATING LICENSE NOS. NPF-37 AND NPF-66

DOCKET NOS. STN 50-454 AND STN 50-455

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

Remove Pages	Insert Pages
3/4 3-1	3/4 3-1
3/4 3-14	3/4 3-14
B 3/4 3-1	B 3/4 3-1
B 3/4 3-2	B 3/4 3-2

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be verified to be within its limit at least once per 18 months. Each verification shall include at least one train such that both trains are verified at least once per 36 months and one channel per function such that all channels are verified at least once every N times 18 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by the performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be verified to be within the limit at least once per 18 months. Each verification shall include at least one train such that both trains are verified at least once per 36 months and one channel per function such that all channels are verified at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation, and 3) sufficient system functions capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

met its allowance. Being that there is a small statisitical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The verification of response time at the specified frequencies provides assurance that the reactor trip and the engineered safety features actuation associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be verified by actual tests in any series of sequential, overlapping or total channel measurements, or by summation of allocated sensor response times with actual tests on the remainder of the channel in any series of sequential or overlapping measurements. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g., vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," provides the basis and methodology for using allocated sensor response times in the overall verification of the Technical Specifications channel response time. The allocations for sensor response times must be verified prior to placing the sensor in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where time response could be affected is replacing the sensing assembly of a transmitter.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss of coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment cooling fans start and automatic valves position, and (11) essential service water pumps start and automatic valves position.

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 76
License No. NPF-72

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 16, 1994, as supplemented by letter dated January 31, 1996, 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. NPF-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 76 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and is to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Ramin R. Assa

Ramin R. Assa, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 26, 1996

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 76
License No. NPF-77

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 16, 1994, as supplemented by letter dated January 31, 1996, 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 1;

   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. NPF-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 76 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and is to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Ramin R. Assa, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 26, 1996

ATTACHMENT TO LICENSE AMENDMENT NOS. 76 AND 76

FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77

DOCKET NOS. STN 50-456 AND STN 50-457

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

Remove Pages	Insert Pages
3/4 3-1	3/4 3-1
3/4 3-14	3/4 3-14
B 3/4 3-2	B 3/4 3-2

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be verified to be within its limit at least once per 18 months. Each verification shall include at least one train such that both trains are verified at least once per 36 months and one channel per function such that all channels are verified at least once every N times 18 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by the performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be verified to be within the limit at least once per 18 months. Each verification shall include at least one train such that both trains are verified at least once per 36 months and one channel per function such that all channels are verified at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statisitical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The verification of response time at the specified frequencies provides assurance that the reactor trip and the engineered safety features actuation associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be verified by actual tests in any series of sequential, overlapping or total channel measurements, or by summation of allocated sensor response times with actual tests on the remainder of the channel in any series of sequential or overlapping measurements. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g., vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," provides the basis and methodology for using allocated sensor response times in the overall verification of the Technical Specifications channel response time. The allocations for sensor response times must be verified prior to placing the sensor in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where time response could be affected is replacing the sensing assembly of a transmitter.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss of coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment cooling fans start and automatic valves position, and (11) essential service water pumps start and automatic valves position.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. NPF-77

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNIT NOS. 1 AND 2

BRAIDWOOD STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-454, STN 50-455, STN 50-456 AND STN 50-457

1.0 INTRODUCTION

By letter dated September 16, 1994, as supplemented by letter dated January 31, 1996, Commonwealth Edison Company (ComEd, the licensee) requested an amendment to Facility Operating License Nos. NPF-37, NPF-66, NPF-72 and NPF-77 to change the Technical Specifications (TS) for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2. The proposed TS changes would eliminate periodic response time testing (RTT) surveillance requirements for the following pressure and differential pressure sensors installed in the specified Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) channels:

1) Barton 764 differential pressure transmitters
   - pressurizer water level (Byron, Units 1 and 2)
   - pressurizer water level (Braidwood, Units 1 and 2)
   - steam generator water level (Byron, Units 1 and 2)
   - steam generator water level (Braidwood, Units 1 and 2)

2) Barton 763 gauge pressure transmitters
   - pressurizer pressure (Byron, Units 1 and 2)
   - steamline pressure (Byron, Units 1 and 2)
   - steamline pressure (Braidwood, Units 1 and 2)
   - wide range pressure (Byron, Units 1 and 2)
   - wide range pressure (Braidwood, Units 1 and 2)

3) Barton 763A gauge pressure transmitters
   - pressurizer pressure (Braidwood, Units 1 and 2)

4) Barton 752 differential pressure transmitters
   - containment pressure (Byron, Units 1 and 2)
   - containment pressure (Braidwood, Units 1 and 2)
   - reactor coolant flow (Byron, Units 1 and 2)
   - reactor coolant flow (Braidwood, Units 1 and 2)
   - refueling water storage tank level (Byron, Units 1 and 2)
   - refueling water storage tank level (Braidwood, Units 1 and 2)

5) Tobar 32PA2 absolute pressure transmitters
   - wide range pressure (Byron, Units 1 and 2)
   - wide range pressure (Braidwood, Units 1 and 2)

Specifically, the proposed TS amendments would revise RTS Instrumentation Surveillance Requirement 4.3.1.2 and ESFAS Instrumentation Surveillance Requirement 4.3.2.2 to indicate that the response time of each function shall be "verified" rather than "tested." The associated Bases section would be revised to indicate that the total channel response time may be verified by either actual response time tests of the entire channel in any series of sequential, overlapping or total channel measurements, or by summation of allocated sensor response times with actual tests on the remainder of the channel in any series of sequential or overlapping measurements. The use of allocated sensor response times would only apply to the specific sensors identified above.

Allocations for specific pressure and differential pressure sensor response times would be obtained from (1) historical records based on acceptable RTT (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g., vendor) test measurements, or (3) vendor engineering specifications. The revised Bases would also indicate that the allocations for the sensor response times must be verified prior to placing the sensor in operational service and re-verified following maintenance that may adversely affect response time, such as replacing the sensing assembly of a transmitter.

2.0 EVALUATION

The licensee noted that Institute of Electrical and Electronic Engineers (IEEE) Standard 338-1977, "Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems," as endorsed by Regulatory Guide 1.118, Revision 2, "Periodic Testing of Electric Power and Protection Systems," dated June 1978, defines a basis for eliminating RTT. Section 6.3.4 of IEEE Standard 338 states in part:

> "Response time testing of all safety-related equipment, per se, is not required if, in lieu of response time testing, the response time of the safety system equipment is verified by functional testing, calibration check, or other tests, or both. This is acceptable if it can be demonstrated that changes in response time

beyond acceptable limits are accompanied by changes in performance characteristics which are detectable during routine periodic tests."

The licensee stated that Westinghouse Topical Report WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," dated August 1995, provides the technical basis for the deletion of periodic RTT of the subject pressure and differential pressure sensors. WCAP-13632-P-A, Revision 2, utilized Electric Power Research Institute (EPRI) failure modes and effects analyses (FMEA) as documented in EPRI Report NP-7243, Revision 1, "Investigation of Response Time Testing Requirements," and Westinghouse Owners' Group (WOG) similarity analyses to justify the elimination of RTT surveillance requirements for numerous types of pressure and differential pressure sensors typically installed in RTS and ESFAS instrumentation loops at Westinghouse plants, including the specific sensors identified in Section 1.0 of this evaluation.

By Safety Evaluation (SE) dated September 5, 1995, the staff approved WCAP-13632-P-A, Revision 2, as a basis for the elimination of TS RTT requirements for each of the pressure sensors identified in WCAP-13632-P-A, Revision 2. As described in the staff's SE, the results of the EPRI FMEAs and the WOG sensor analyses indicated that, in general, potential sensor component failure modes associated with sensors identified in WCAP-13632-P-A, Revision 2, would not affect sensor response time independently of sensor output. Therefore, sensor failure modes that have the potential to affect sensor response time would be detected during the performance of other TS surveillance tests, such as channel checks and calibrations. Based on this information, the staff concluded that RTT is, in general, redundant to other TS surveillance requirements.

However, the EPRI results did identify several potential failure modes in certain pressure sensors that could affect sensor response time without concurrently affecting sensor output. To address these failure modes and other generic concerns, the staff stipulated four actions that licensees were to commit to take, if applicable, when eliminating sensor RTT. The licensee satisfactorily addressed the four actions. First, the staff's September 5, 1995, SE specified that licensees were to perform a hydraulic RTT prior to installation of a new transmitter/switch or following refurbishment of the transmitter/switch to determine an initial sensor-specific response time value. In response to this action, ComEd committed to revise or develop applicable station and corporate procedures and instructions to ensure that this RTT is performed, as required. In addition, upon completing the RTT of the newly installed or refurbished transmitter, ComEd will verify that the associated total channel response time is less than the value specified in the updated final safety analysis report by summing the transmitter RTT result with the most recent RTT results for the remaining channel components. The staff finds this commitment acceptable.

Secondly, the EPRI FMEAs identified crimped capillaries as a manufacturing/handling defect that has the potential to affect response times of sensors containing capillaries. As a result, the staff specified that for transmitters and switches with capillary tubes, a RTT is to be performed after initial installation and after any maintenance or modification activity that could damage the capillary tubes. In response to this action, ComEd committed to revise or develop appropriate station and corporate procedures and instructions to stipulate that transmitters and switches utilizing capillary tubes must be subjected to RTT after initial installation and following any maintenance or modification activity which could damage the capillary tubes. The licensee noted that a RTT would not be performed after any routine calibrations or unscheduled calibrations that do not adversely affect the capillary tubes. In addition, upon completing the RTT of the newly installed or modified transmitter, ComEd will verify that the associated total channel response time is less than the value specified in the updated final safety analysis report by summing the transmitter RTT result with the most recent RTT results for the remaining channel components. The staff finds this commitment acceptable.

The third and fourth stipulated actions in the staff's SE were included as a result of identified failure modes associated with transmitters that have variable damping potentiometers and Rosemount pressure and differential pressure transmitters, respectively. However, these two actions are not applicable to the Byron and Braidwood plants because there are no variable damping transmitters or Rosemount transmitters installed in any RTS or ESFAS application for which RTT is required.

The licensee proposed using allocated sensor response times in accordance with the methodology contained in Section 9.0 of WCAP-13632-P-A, Revision 2, to verify total RTS or ESFAS channel response time. Allocations for sensor response times would be obtained from (1) historical records based on acceptable RTT (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g., vendor) test measurements, or (3) vendor engineering specifications. There is no specific recommendation regarding which of these methods to use, although the value should be increasingly more conservative progressing through these methods. Available manufacturer supplied and Westinghouse engineering specification response time values for the subject pressure sensors are shown in Table 9-1 of WCAP-13632-P-A, Revision 2. The total channel response time is obtained by summing the allocated sensor response time with the measured response time of the remainder of the channel. This methodology is described in WCAP-13632-P-A, Revision 2, and was previously approved in the staff's generic SE dated September 5, 1995. Reference to this methodology by ComEd is, therefore, acceptable.

3.0 CONCLUSION

To meet the guidance of Regulatory Guide 1.118, Revision 2, and IEEE 338-1977, Section 6.3.4, RTT is needed unless it is shown that changes in the response time will be accompanied by changes in performance characteristics which are detectable during routine periodic surveillance tests. The sensor analyses results as described in WCAP-13632-P-A, Revision 2, concluded that RTT is redundant to other periodic surveillance tests, such as channel checks and calibrations, because these other surveillance tests will detect sensor component failures that cause response time degradation.

Based on the licensee's adoption of WCAP-13632-P-A, Revision 2 and satisfactorily addressing the four actions listed in the staff's SE of September 5, 1995, approving WCAP-13632-P-A, Revision 2, the staff concludes that (1) other existing TS surveillance requirements for the subject pressure and differential pressure sensors provide confidence that the safety function of the plant instrumentation will be satisfied without the need for specific RTT, and (2) plant specific actions will be taken as appropriate when replacing/refurbishing transmitters and when handling transmitters with capillary tubes. The staff, therefore, concludes that ComEd's proposal to eliminate the TS RTT requirements for the pressure and differential pressure sensors identified in Section 1.0 of this evaluation is acceptable.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change survillance 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 (61 FR 10393). 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.

6.0 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.

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