Mr. Charles Pardee
Chief Nuclear Officer
and Senior Vice President
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

January 29, 2008

SUBJECT: BYRON STATION, UNIT NOS. 1 AND 2, AND BRAIDWOOD STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: TECHNICAL SPECIFICATION REQUEST TO EXTEND REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM COMPLETION TIMES, BYPASS TEST TIMES, AND SURVEILLANCE TEST INTERVALS (TAC NOS. MD4009, MD4010, MD4011, AND MD4012)

Dear Mr. Pardee:

The Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 153 to Facility Operating License No. NPF-37 and Amendment No. 153 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 148 to Facility Operating License No. NPF-72 and Amendment No. 148 to Facility Operating License No. NPF-77 for the Braidwood Station, Units 1 and 2, respectively.

The amendments are in response to your application dated January 8, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML080180213), as supplemented by letter dated October 12, 2007 (ADAMS Accession No. ML072880107). The amendments extend the reactor trip system (RTS) and engineered safety features actuation system (ESFAS) completion times, bypass test times, and surveillance test intervals for technical specifications (TS) 3.3.1, “RTS Instrumentation,” TS 3.3.2, “ESFAS Instrumentation,” and TS 3.3.6, “Containment Ventilation Isolation Instrumentation.”

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,

/RA/

Meghan M. Thorpe-Kavanaugh, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 153 to NPF-37
2. Amendment No. 153 to NPF-66
3. Amendment No. 148 to NPF-72
4. Amendment No. 148 to NPF-77
5. Safety Evaluation

cc w/encls: See next page

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,

/RA/

Meghan M. Thorpe-Kavanaugh, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 153 to NPF-37
2. Amendment No. 153 to NPF-66
3. Amendment No. 148 to NPF-72
4. Amendment No. 148 to NPF-77
5. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION:

PUBLIC
RidsRgn3MailCenter
RidsNrrLAEWhitt
GHill (8)
RidsNrrDssScvb
RidsNrrDirsItsb
RidsNrrDorlLpl3-2
RidsAcrsAcnwMailCenter
RidsNrrDorlDpr
LPL3-2 R/F
RidsOgcRp
RidsNrrPMMThorpe-Kavanaugh
RidsNrrDraApla

Package: ML080110168
Amendment: ML080110179
Tech Spec Pages: ML080370155 *By SE dated 12/13/08 **By SE dated 10/11/07 NRR-058

OFFICE	LPL3-2/PM	LPL3-2/LA	DRA/APLA/BC	DSS/SCVB/BC	DIRS/ITSB/BC	OGC	LPL3-2/BC
NAME	MThorpe-Kavanaugh	EWhitt	MRubin*	RDennig**	TKobetz	MSmith NLO wi/comments	RGibbs
DATE	1/29/08	1/29/08	12 / 13 / 08	10 / 11 / 07	01 / 18 /08	01/28/08	1/29/2008

OFFICIAL RECORD COPY

Byron/Braidwood Stations

cc:

Corporate DistributionExelon Generation
Company, LLC
via e-mail

Braidwood Distribution
Exelon Generation Company, LLC
via e-mail

Byron Distribution
Exelon Generation Company, LLC
via e-mail

Dwain W. Alexander, Project Manager
Westinghouse Electric Company
via e-mail

Howard A. Learner
Environmental Law and Policy
Center of the Midwest
via e-mail

Byron Senior Resident Inspector
U.S. Nuclear Regulatory Commission
via e-mail

Byron Resident Inspectors
U.S. Nuclear Regulatory Commission
via e-mail

Ms. Lorraine Creek
RR 1, Box 182
Manteno, IL 60950

Chairman, Ogle County Board
Post Office Box 357
Oregon, IL 61061

Mrs. Phillip B. Johnson
1907 Stratford Lane
Rockford, IL 61107

Attorney General
500 S. Second Street
Springfield, IL 62701

Illinois Emergency Management Agency
Division of Disaster Assistance &
Preparedness
via e-mail

Braidwood Senior Resident Inspector
U.S. Nuclear Regulatory Commission
via e-mail

County Executive
via e-mail

County Executive Secretary
via e-mail

Ms. Bridget Little Rorem
Appleseed Coordinator
via e-mail

Chairman
Will County Board of Supervisors
Will County Board Courthouse
Joliet, IL 60434

Mr. Barry Quigley
3512 Louisiana
Rockford, IL 61108

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 153
License No. NPF-37

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated January 8, 2007, as supplemented by letter dated October 12, 2007, 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. 153 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 shall be implemented within 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/
Russell Gibbs, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: January 29, 2008

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 153
License No. NPF-66

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated January 8, 2007, as supplemented by letter dated October 12, 2007, 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-66 is hereby amended to read as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 153 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. 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 shall be implemented within 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Russell Gibbs, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: January 29, 2008

ATTACHMENT TO LICENSE AMENDMENT NOS. 153 AND 153

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

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

Replace the following pages of the Facility Operating License and Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License NPF-37	License NPF-37
Page 3	Page 3
License NPF-66	License NPF-66
Page 3	Page 3
TSs	TSs
3.3.1-2	3.3.1-2
3.3.1-3	3.3.1-3
3.3.1-4	3.3.1-4
3.3.1-5	3.3.1-5
3.3.1-6	3.3.1-6
3.3.1-7	3.3.1-7
3.3.1-8	3.3.1-8
3.3.1-9	3.3.1-9
3.3.1-10	3.3.1-10
3.3.1-14	3.3.1-14
3.3.2-2	3.3.2-2
3.3.2-3	3.3.2-3
3.3.2-4	3.3.2-4
3.3.2-5	3.3.2-5
3.3.2-7	3.3.2-7
3.3.6-3	3.3.6-3

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulation set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3586.6 megawatts thermal (100 percent power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 153 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) Deleted.

(4) Deleted.

(5) Deleted.

(6) The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the licensee's Fire Protection Report, and as approved in the SER dated February 1987 through Supplement No. 8, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulation set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3586.6 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 153, 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. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Deleted.

(4) Deleted.

(5) Deleted.

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 148
License No. NPF-72

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated January 8, 2007, as supplemented by letter dated October 12, 2007, 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. 148 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 shall be implemented within 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Russell Gibbs, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: January 29, 2008

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 148
License No. NPF-77

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated January 8, 2007, as supplemented by letter dated October 12, 2007, 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-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 148 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 if its issuance and shall be implemented within 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Russell Gibbs, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: January 29, 2008

ATTACHMENT TO LICENSE AMENDMENT NOS. 148 AND 148

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

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

Replace the following pages of the Facility Operating Licenses and Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License NPF-72	License NPF-72
Page 3	Page 3
License NPF-77	License NPF-77
Page 3	Page 3
TSs	TSs
3.3.1-2	3.3.1-2
3.3.1-3	3.3.1-3
3.3.1-4	3.3.1-4
3.3.1-5	3.3.1-5
3.3.1-6	3.3.1-6
3.3.1-7	3.3.1-7
3.3.1-8	3.3.1-8
3.3.1-9	3.3.1-9
3.3.1-10	3.3.1-10
3.3.1-14	3.3.1-14
3.3.2-2	3.3.2-2
3.3.2-3	3.3.2-3
3.3.2-4	3.3.2-4
3.3.2-5	3.3.2-5
3.3.2-7	3.3.2-7
3.3.6-3	3.3.6-3

(3) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at reactor core power levels is not in excess of 3586.6 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein and other items identified in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A as revised through Amendment No. 148, 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) <u>Emergency Planning</u>

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Exelon Generation Company, LLC pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts are required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Exelon Generation Company, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels is not in excess of 3586.6 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein and other items identified in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 148, and the Environmental Protection Plan contained in Appendix B, both of which are 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) Emergency Planning

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 153 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 153 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 148 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 148 TO FACILITY OPERATING LICENSE NO. NPF-77

EXELON GENERATION COMPANY, LLC

BYRON STATION, UNIT NOS. 1 AND 2

BRAIDWOOD STATION, UNITS 1 AND 2

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

STN 50-456 AND STN 50-457

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC, the Commission) dated January 8, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML080180213), as supplemented by letter dated October 12, 2007 (ADAMS Accession No. ML072880107). Exelon Generation Company, LLC (the licensee) requested changes to the technical specifications (TSs) facility operating licenses, surveillance requirements (SRs) for the Byron Station, Unit Nos. 1 and 2 (Byron), and the Braidwood Station, Units 1 and 2 (Braidwood), in accordance with Title 10 of *the Code of Federal Regulations* (10 CFR), Section 50.90, "Application for amendment of license or construction permit." The supplement provided additional information that clarified the application, but did not expand the scope of the application as originally noticed and did not change the NRC staff's original proposed no-significant-hazards consideration (NSHC) determination as published in the *Federal Register* on March 27, 2007 (72 FR 14305).

The proposed amendment adopts changes approved by the staff in the Westinghouse Commercial Atomic Power (WCAP) topical report WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the RPS [reactor protection system] and ESFAS [engineered safety features actuation system] Test Times and Completion Times," issued October 1998, as approved by the NRC in a letter dated July 15, 1998. Implementation of the proposed changes is in accordance with Technical Specification Task Force (TSTF) Change Traveler TSTF-418, Revision 2, "RPS and ESFAS Test Times and Completion Times (WCAP-14333)." The NRC approved TSTF-418 by letter dated April 2, 2003.

Enclosure

In addition, the proposed amendment adopts changes approved by the staff in WCAP-15376-P-A, Revision 1, “Risk-Informed Assessment of the RTS [Reactor Trip System] and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times,” dated March 2003, as approved by the NRC in a letter dated December 20, 2002. Implementation of the proposed changes is in accordance with TSTF-411, Revision 1, “Surveillance Test Interval Extension for Components of the Reactor Protection System (WCAP-15376).” The NRC approved TSTF-411, Revision 1, by letter dated August 30, 2002.

The proposed change revises Byron and Braidwood completion times (CTs), bypass test times, and surveillance test intervals (STIs) for TS 3.3.1, “RTS Instrumentation,” TS 3.3.2, “ESFAS Instrumentation,” and TS 3.3.6, “Containment Ventilation Isolation Instrumentation.” Attachment 1, Section 3.0, “Proposed Changes,” of the licensee’s submittal describes the specific TS changes. The licensee also included revised Byron and Braidwood TS Bases in Attachments 4A and 4B.

The topical reports state that the proposed changes to CTs, bypass test times, and STIs will allow additional time to perform tests and maintenance, enhance safety, provide additional operational flexibility, and reduce the potential for forced outages related to compliance with the reactor protection system RPS and ESFAS instrumentation TS.

1.1 Related NRC Actions

This license amendment is not related to or in response to any ongoing NRC activities (e.g., generic letters (GLs)).

1.2 Background

The Pressurized-Water Reactor Owners Group (PWROG), formerly the Westinghouse Owners Group, Technical Specifications Optimization Program (TOP) evaluated changes to STIs and CTs (allowed outage time) for the analog channels, logic cabinets, master and slave relays, and reactor trip breakers (RTBs). The methodology evaluated increases in surveillance intervals, test and maintenance out-of-service times, and the bypassing of portions of the RPS during test and maintenance. In 1983, PWROG submitted WCAP-10271-P, “Evaluation of Surveillance Frequencies and Out-of-Service Times for the Reactor Protection Instrumentation System,” which provided a methodology for justifying revisions to a plant’s RPS TS. PWROG stated in WCAP-10271 that plant staff devoted significant time and effort to perform, review, document, and track surveillance activities that, in many instances, may not be necessary because of the high reliability of the equipment. Part of the justification for the changes was their anticipated small impact on plant risk.

By letter dated February 21, 1985, the NRC accepted WCAP-10271, including Supplement 1, with conditions. In 1989, the NRC staff issued a safety evaluation report (SER) for WCAP-10271, Supplement 2, which approved similar relaxations for the ESFAS. An additional supplemental SER issued in 1990 provided consistency between RTS and ESFAS STIs and CTs. The NRC subsequently adopted the TS changes proposed by WCAP-10271 into NUREG-1431, “Standard Technical Specifications Westinghouse Plants,” Revision 0, issued September 1992. Byron and Braidwood previously implemented WCAP-10271 in License Amendments 55 and 44, respectively.

After the approval of WCAP-10271 and its supplements, the PWROG submitted WCAP-14333-P in May 1995. The purpose of this topical report was to provide justification for the following TS relaxations beyond those approved in WCAP-10271:

- increase the bypass test times and CTs for both the solid-state and relay protection system RTS and ESFAS designs for the analog channels, increase the CT from 6 hours to 72 hours and the bypass test time from 4 hours to 12 hours and for the logic cabinets, master relays, and slave relays, increase the CT from 6 hours to 24 hours.

- for cases in which the logic cabinet and RTB both cause their train to be inoperable when in test or maintenance, allow bypassing of the RTB for the period of time equivalent to the bypass test time for the logic cabinets, provided that both are tested at the same time and the plant design is such that both the RTB and the logic cabinet cause their associated electrical trains to be inoperable during test or maintenance.

The NRC staff accepted WCAP-14333 by letter dated July 15, 1998. Following the approval of WCAP-14333, PWROG submitted WCAP-15376 to the NRC staff on November 8, 2000, which the NRC staff subsequently approved by letter dated December 20, 2002.

WCAP-15376 specifically evaluated the analog channels, logic cabinets, master relays, and RTBs. WCAP-15376 evaluated both the solid-state protection system (SSPS) and the relay protection system. WCAP-15376 provided justification for the following TS relaxations:

- additional extension of the STIs for components of the RPS and ESFAS to those previously approved in WCAP-10271

- extension of the STI, CT, and bypass test times for the RTBs

Attachment 1, Section 3.0, of the licensee's submittal lists the specific TS changes proposed with the implementation of WCAP-14333 and WCAP-15376.

2.0 REGULATORY EVALUATION

The proposed TS modifications affect the RPS (i.e., RTS and ESFAS). The RTS is designed to initiate a reactor trip when the system exceeds limits to permissible operation. The ESFAS is designed to actuate emergency systems for accidents that challenge the normal control and heat removal systems.

The RPS comprises several major functions, including nuclear and process instrumentation, logic, reactor trip, and ESFAS actuation. Instrumentation includes sensors, power supplies, signal processing, and bistable outputs and typically consists of three or four channels. Instrumentation signals (i.e., bistable outputs) feed relays that input into the logic portion of the RPS. The logic (i.e., logic cabinets) includes two redundant and independent logic blocks consisting of two trains (A and B) of RPS logic where the input coincidence for various trip functions is determined. Either logic train initiates the ESFAS function through master and slave relays.

In addition, the RPS includes actuation paths from the Train A and Train B RPS logic to the RTBs. Normally, an RTB receives its signal from its associated RPS logic train. The system has bypass breakers for when a breaker is out of service. In this configuration, the bypass breaker is associated with the logic train of the operable RTB. The RPS utilizes two normally closed RTBs and two normally open bypass breakers. Train A RPS logic actuates RTB A, and Train B logic actuates RTB B. Opening of either RTB will disconnect power from the control rods, causing a reactor trip.

Both Byron and Braidwood utilize an SSPS for the logic portion of the RPS.

2.1 Applicable Regulations

This section discusses the regulations applicable to this license amendment request (LAR).

Pursuant to 10 CFR 50.36, "Technical specifications," a licensee's TSs must have surveillance requirements (SRs) relating to test, calibration, or inspection to ensure that the facility maintains the necessary quality of systems and components, facility operations are within safety limits, and facility equipment will meet the limiting conditions for operation (LCO). The SRs may include mode restrictions based on the safety aspects of conducting the surveillance in excluded modes.

Although 10 CFR 50.36 does not list specific TS requirements, implicit within this rule are the requirements that action be taken when an LCO is not being met and that the SRs, bypass test times, and CTs specified in the TSs be based on reasonable protection of the public health and safety. Therefore, the NRC staff must be able to conclude that there is reasonable assurance that the RTS/ESFAS functions affected by these proposed TS changes will perform their required safety functions in accordance with the design-basis accidents described in Chapter 15 of the licensee's final safety analysis report, based on the proposed SRs, bypass test times, and CTs. As set forth in 10 CFR 50.36, a licensee's TSs must establish the LCO and contain certain information. This requirement includes SRs, CTs, and bypass test times for structures, systems, and components (SSCs) required for safe operation of the facility, such as the RTS and ESFAS.

The Maintenance Rule, 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," requires licensees to monitor the performance or condition of SSCs against licensee-established goals in a manner sufficient to provide reasonable assurance that SSCs are capable of fulfilling their intended functions. The implementation and monitoring program guidance of Section 2.3 of Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," issued November 2002, and Section 3 of RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," issued August 1998, states that monitoring performed in conformance with the Maintenance Rule can be used when it is sufficient for the SSCs affected by the risk-informed application. In addition, Maintenance Rule (Section (a)(4)), as it relates to the proposed surveillance, bypass test times, and CTs, requires the assessment and management of the increase in risk that may result from the proposed maintenance activity.

According to General Design Criterion (GDC) 13, "Instrumentation and Control," the licensee shall provide appropriate controls to maintain these variables and systems within prescribed operating ranges.

According to GDC 21, “Protection System Reliability and Testability,” the design of the protection system shall provide for high functional reliability and inservice testability commensurate with the safety functions to be performed. The design of the protection system shall permit periodic testing of its functioning when the reactor is in operation, including the capability to test channels independently to determine failures and losses of redundancy that may have occurred. Finally, 10 CFR 50.90 addresses the requirements for a licensee to amend its license, including the TSs.

Attachment 1, Section 5.2, of the licensee’s submittal references additional regulatory requirements and criteria applicable to the licensee’s implementation of WCAP-14333 and WCAP-15376.

2.2 Applicable Regulatory Criteria/Guidelines

RG 1.174 describes a risk-informed approach with associated acceptance guidelines for licensees to assess the nature and impact of proposed permanent licensing basis changes by considering engineering issues and applying risk insights.

RG 1.177 describes an acceptable risk-informed approach and additional acceptance guidance geared toward the assessment of proposed permanent TS CT changes. RG 1.177 identifies a three-tiered approach for the licensee’s evaluation of the risk associated with a proposed CT TS change, as discussed below:

- Tier 1 assesses the risk impact of the proposed change in accordance with acceptance guidelines consistent with the Commission’s Safety Goal Policy Statement, as documented in RGs 1.174 and 1.177. The first tier assesses the impact on operational plant risk based on the change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF). It also evaluates plant risk while equipment covered by the proposed CT is out of service, as represented by incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP). Tier 1 also addresses probabilistic risk assessment (PRA) quality, including the technical adequacy of the licensee’s plant-specific PRA for the subject application. Tier 1 also considers the cumulative risk of the present TS change in light of past (related) applications or additional applications under review along with uncertainty/sensitivity analysis with respect to the assumptions related to the proposed TS change.

- Tier 2 identifies and evaluates any potential risk-significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed license amendment, is taken out of service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The purpose of this evaluation is to ensure that appropriate restrictions are in place such that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed CT is implemented.

- Tier 3 addresses the licensee’s overall configuration risk management program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk-significant plant configurations resulting from maintenance or other operational activities and that the licensee takes appropriate compensatory measures to avoid risk-significant configurations that may not have been considered during the Tier 2 evaluation. Compared with Tier 2, Tier 3 provides additional coverage to ensure that the licensee identifies risk-significant plant equipment outage configurations in a timely manner and appropriately evaluates the risk impact of out-of-service equipment before performing any maintenance activity over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule (Section (a)(4)), which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance testing and corrective and preventive maintenance, subject to the guidance provided in RG 1.177, Section 2.3.7.1, and the adequacy of the licensee's program and PRA model for this application. The purpose of the CRMP is to ensure that the licensee will appropriately assess from a risk perspective equipment removed from service before or during the proposed extended CT.

RGs 1.174 and 1.177 also describe acceptable implementation strategies and performance monitoring plans to help ensure that the assumptions and analyses used to support the proposed TS changes will remain valid. The monitoring program should include a means to adequately track the performance of equipment that, when degraded, can affect the conclusions of the licensee's evaluation for the proposed licensing basis change. RG 1.174 states that monitoring performed in accordance with the Maintenance Rule can be used when such monitoring is sufficient for the SSCs affected by the risk-informed application.

Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP), provides general guidance for evaluating the technical basis for proposed risk-informed changes. SRP Section 16.1, "Risk-Informed Decision Making: Technical Specifications," provides more specific guidance related to risk-informed TS changes, including CT changes as part of risk-informed decision making. SRP Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," addresses the technical adequacy of a baseline PRA used by a licensee to support license amendments for an operating reactor. SRP Section 19.2 states that a risk-informed application should be evaluated to ensure that the proposed changes meet the following five key principles:

(1) The proposed change meets the current regulations, unless it explicitly relates to a requested exemption or rule change.

(2) The proposed change is consistent with the defense-in-depth philosophy.

(3) The proposed change maintains sufficient safety margins.

(4) When proposed changes increase CDF or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

(5) The licensee should monitor the impact of the proposed change using performance measurement strategies.

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's analyses in support of its proposed license amendment, which are described in the submittal dated January 8, 2007, as supplemented by letter dated October 12, 2007.

3.1 Detailed Description of the Proposed Change

The following table summarizes the proposed WCAP-14333 changes, as applicable to Byron and Braidwood.

RPS/ESFAS Components	CT		Bypass Test Time	
	Current (Hour)	Proposed (Hour)	Current (Hour)	Proposed (Hour)
Analog Channels	6+6[1]	72+6	4	12
Logic Cabinets	6+6	24+6	4	No Change
Master Relays	6+6	24+6	4	No Change
Slave Relays	6+6	24+6	4	No Change
RTB	6	No Change[2]	2	No Change[2]

1. The +6 hours is the time allowed for the specified mode change.

2. WCAP-14333 does not directly revise the RTB CT and bypass test times, and it is assumed that the bypass test times for the RTBs and the logic cabinets are separate and independent. However, WCAP-14333 assumes that with either a logic cabinet or RTB in test or maintenance their associated train is also unavailable. Based on this, the analysis presented in WCAP-14333 includes a provision to accept a bypass test time of the RTBs equivalent to the bypass test time for the logic cabinets provided that (1) both are tested concurrently, and (2) the plant design is such that both the RTB and the logic cabinet cause their associated electrical trains to be inoperable during test or maintenance. Therefore, the RTB bypass test time is extended to 4 hours for this maintenance configuration. With the implementation of WCAP-15376, the RTB bypass test time is increased to 4 hours consistent with the logic cabinet.

The following table summarizes the proposed WCAP-15376 changes, as applicable to Byron and Braidwood.

RPS Component	STI		CT		Bypass Test Time	
	Current (Month)	Proposed (Month)	Current (Hour)	Proposed (Hour)	Current (Hour)	Proposed (Hour)
Logic Cabinets	2	6	No Change Requested		No Change Requested	
Master Relays	2	6				
Analog Channels	3	6				
RTB	2	4	1	24	2	4

3.2 Review of Methodology

In accordance with SRP Sections 19.1, 19.2, and 16.1, the staff reviewed the Byron and Braidwood incorporation of WCAP-14333 and WCAP-15376 using the three-tiered approach and the five key principles of risk-informed decisionmaking presented in RGs 1.174 and 1.177 and the SER conditions and limitations for WCAP-14333 and WCAP-15376.

3.3 Key Information Used in the Review

The key information used in the NRC staff's review comes from Attachments 1, 5, and 6 of the licensee's LAR dated January 8, 2007, as supplemented by the request for additional information (RAI) responses dated October 12, 2007; TSTF-411 and TSTF-418; and the WCAP-14333 and WCAP-15376 NRC staff's SERs. The NRC staff also referred to previous SERs related to WCAP-10271 and the licensee's individual plant examination (IPE) and individual plant examination of external events (IPEEE) assessments.

3.4 Comparison to Regulatory Criteria/Guidelines

The following sections present the NRC staff's evaluation of the licensee's proposed amendment to extend CTs and bypass test times using the three-tiered approach and the key five principles outlined in RGs 1.174 and 1.177.

3.4.1 Traditional Engineering Evaluation

The traditional engineering evaluation addresses key principles 1, 2, 3, and 5 of the NRC staff's philosophy of risk-informed decision making, which concern compliance with current regulations, evaluation of defense in depth, evaluation of safety margins, and performance measurement strategies. The NRC staff previously performed a generic evaluation of WCAP-15376. The NRC staff's review of the changes found that WCAP-15376 was consistent with the accepted guidelines of RG 1.174 and RG 1.177, and NRC staff guidance as outlined in NUREG-0800, "Standard Review Plan." From traditional engineering insights, the NRC staff found that the proposed changes in WCAP-15376 continue to meet the regulations, have no impact on the defense-in-depth philosophy, and would not involve a significant reduction in the margin of safety.

RGs 1.174 and 1.177 also establish the need for an implementation and monitoring program to ensure that extensions to TS CTs, bypass test times, and surveillance intervals do not degrade operational safety over time and that no adverse degradation results from unanticipated degradation or common-cause mechanisms. The purpose of an implementation and monitoring program is to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of SSCs impacted by the change. RG 1.174 states that monitoring performed in conformance with the Maintenance Rule can be used when such monitoring is sufficient for the SSCs affected by the risk-informed application. Section 3.4.3 of this safety evaluation (SE) provides the staff's evaluation of the licensee's implementation and monitoring program.

3.4.2 Risk Evaluation

The probabilistic risk assessment (PRA) evaluation addresses key principle 4 of the NRC staff's philosophy of risk-informed decisionmaking. The changes proposed by the licensee employ a risk-informed approach to justify changes to CTs, bypass test times, and STIs. The risk metrics, ΔCDF, ΔLERF, ICCDP, and ICLERP, developed in the topical report and that the licensee used to evaluate the impact of the proposed changes are consistent with those presented in RGs 1.174 and 1.177.

3.4.2.1 Applicability of WCAP-14333 and WCAP-15376 to Byron/Braidwood

To determine that WCAP-14333 and WCAP-15736 are applicable to Byron and Braidwood, the licensee addressed the conditions and limitations of the NRC staff's SERs and the implementation guidance developed by PWROG that compares plant-specific data to the generic analysis assumptions. The evaluation compared the general baseline assumptions, including surveillance, maintenance, calibration, actuation signals, procedures, and operator actions, to confirm that the generic evaluation assumptions used in the topical reports are also applicable to Byron and Braidwood.

The following paragraphs discuss the licensee's evaluation of the SE conditions and limitations of WCAP-14333 and WCAP-15376.

(1) A licensee should confirm the applicability of the WCAP-14333 and WCAP-15376 analyses for its plant.

    Section 4.4.3 of Attachment 1 and Tables 1 through 5 of Attachment 6 of the licensee's submittal provide the evaluation for WCAP-14333 and WCAP-15376. The evaluation included a comparison of parameters and assumptions with Byron and Braidwood plant-specific data. Data included plant-specific signals, component test and maintenance intervals, procedures, and anticipated transient without scram (ATWS) information. The licensee also evaluated containment failure modes.

    Based on the evaluation presented in Section 3.4.2.1, Tier 1, of this SE, the NRC staff considers the condition satisfied for Byron and Braidwood.

(2) Under WCAP-14333, the licensee should address the Tier 2 and Tier 3 analyses, including CRMP insights, by confirming that these insights are incorporated into its decision-making process before taking equipment out of service.

    Under WCAP-15376, the licensee should address the Tier 2 and Tier 3 analysis, including risk-significant configuration insights, and confirm that these insights are incorporated into the plant-specific CRMP.

    Based on the evaluation presented in Section 3.4.2.2 (Tier 2) and Section 3.4.2.3 (Tier 3) of this SE, the NRC staff considers these conditions satisfied for Byron and Braidwood.

(3) The licensee should evaluate the risk impact of concurrent testing of one logic cabinet and associated RTB on a plant-specific basis to ensure conformance with the WCAP-15376 evaluation, including the guidance of RGs 1.174 and 1.177.

The licensee showed that the generic analysis presented in WCAP-15376 is applicable to Byron and Braidwood. WCAP-15376 did not specifically evaluate or preclude concurrent testing of one logic cabinet and associated RTB. Based on this, the NRC staff questioned the applicability of the topical report to this particular maintenance configuration. In response to an NRC staff's RAI on WCAP-15376, PWROG provided risk estimates for this more limiting configuration. The resulting generic ICCDP estimate was within the acceptance guidelines of RG 1.177. Based on the applicability of WCAP-15376 to Byron and Braidwood and an ICCDP estimate within the acceptance guidelines of RG 1.177, the NRC staff considers condition 3 to be satisfied.

(4) To ensure consistency with the reference plant, the licensee should confirm that the model assumptions for human reliability in WCAP-15376 are applicable to the plant-specific configuration.

The licensee confirmed that the assumptions regarding human reliability used in WCAP-15376 are applicable to Byron and Braidwood. This review concluded that for the operator actions identified in WCAP-15376, plant procedures are available consistent with the assumptions in WCAP-15376. Attachment 6 to the licensee's submittal noted that the topical report recognized operator action to achieve reactor trip by interrupting power to the motor-generator sets. Byron and Braidwood do not have the means to directly disconnect the motor-generator set from the control rod drives, and the PRA does not credit this. Still, the ATWS contribution to the CDF is lower than that of the WCAP-15376 reference plant. Based on the above, the NRC staff considers condition 4 to be satisfied.

(5) For future digital upgrades with increased scope, integration, and architectural differences beyond those of Eagle 21, the NRC staff finds that the generic applicability of WCAP-15376 to a future digital system is not clear and should be considered on a plant-specific basis.

Because the Byron and Braidwood ESFAS design is based on the SSPS and do not employ the Eagle 21 system, this condition is not applicable to the implementation of WCAP-15376 at these units.

(6) WCAP-15376 included an additional condition based on the PWROG response to an NRC staff's RAI that committed each plant to review its plant-specific setpoint calculation methodology to ensure that the extended STIs do not adversely impact the plant-specific setpoint calculations and assumptions for instrumentation associated with the extended STIs.

The licensee reviewed the plant specific RTS and ESFAS setpoint uncertainty calculations and assumptions, including instrument drift, to determine the impact of extending the channel operational test (COT) surveillance from 92 days to 184 days and determined that the values used in the Byron and Braidwood setpoint studies properly accounted for drift due to the extended STIs. Based on their review of the setpoint uncertainty calculations, the licensee stated that they do not anticipate any impact on setpoint uncertainty due to extending the STIs from 92 days to 184 days. However, the licensee committed to trend and evaluate as-found and as-left data for the three representative trip functions analyzed in WCAP-15376 (i.e., over-temperature ΔT (OTDT), steam generator (SG) level, and pressurizer pressure) for two years (4 data points) following implementation of the proposed changes.

The NRC staff review found the proposed change to extend COT surveillance from 92 days to 184 days in Byron and Braidwood TSs sections 3.3.1, 3.3.2 within the scope of the NRC staff SE on WCAP-15376 and acceptable. Further to this, the NRC staff believes that the licensee's commitment to implement the proposed administrative control within 120 days of NRC approval of the LAR, will ensure SSPS availability.

3.4.2.1.1 Tier 1 - Probabilistic Risk Assessment Capability and Insights

The first tier evaluates the impact of the proposed changes on plant operational risk based on the Byron and Braidwood's implementation of WCAP-14333 and WCAP-15376. The Tier 1 of the NRC staff review involves (I) evaluation of the validity of the PRA and its application to the proposed changes, and (II) evaluation of the PRA results and insights based on the licensee's proposed application.

(I) PRA Technical Adequacy

PRA Quality

The objective of the PRA technical adequacy review is to determine whether WCAP-14333 and WCAP-15376, which are used in evaluating the proposed RTS and ESFAS CT, test bypass time, and STI extensions, are of sufficient scope and detail for this application. WCAP-14333 and WCAP-15376 provided a generic PRA model for the evaluation of the CT, test bypass time and STI extensions. The NRC staff found this generic model and the WCAP-14333/WCAP-15376 evaluations to be acceptable on a generic basis in SERs dated July 15, 1998, and December 20, 2002, respectively. Although the SERs accepted the use of a representative model as generally reasonable, the application of the representative model and the associated results to a specific plant introduces a degree of uncertainty because of modeling, design, and operational differences. Therefore, each licensee adopting WCAP-14333 will need to confirm that the WCAP-14333 and WCAP-15376 analysis and results are applicable to its plant.

The NRC staff reviewed the information provided in the proposed LAR and the findings and conditions of the NRC staff WCAP-14333 and WCAP-15376 SER. The WCAP-14333 and WCAP-15376 conditions and limitations identified by the NRC staff were considered limiting for Byron and Braidwood in that the topical reports do not specify the use of the Byron and Braidwood or plant-specific estimates of ΔCDF, ΔLERF, ICCDP, or ICLERP in the implementation of either topical report. However, in its SER for WCAP-14333 and WCAP-15376, the staff found that the applicability of the generic PRA analysis for the proposed CT, bypass test time, and STI changes to other Westinghouse plants may not be representative based on design variations in actuated systems and the contribution to plant risk from accident classes impacted by the proposed change. The NRC staff therefore concluded that each licensee would need to address any differences between its plant and the representative plant that could increase the CT, bypass test time, or STI risk significance. The licensee reviewed the scope and detail of the Byron and Braidwood PRA using the representative topical report PRA parameters to demonstrate the plant-specific applicability of the proposed CT, bypass test times, and STI changes. The licensee compared actuation logic; component test, maintenance, and calibration times/intervals; at-power maintenance; ATWS; total internal events CDF; transient events; operator actions; RTS trip actuation signals; and ESFAS actuation signals to plant specific values. Based on the comparison in accordance with the implementation guidelines for

WCAP-14333 and WCAP-15376, the licensee concluded that WCAP-14333 is applicable to Byron and Braidwood.

In its RAI response, the licensee stated that it has approved a revised PRA model for Byron and Braidwood from that used in the initial evaluation of WCAP-14333 and WCAP-15376. The licensee confirmed that Byron and Braidwood conformance to the WCAP-14333 and WCAP-15376 implementation criteria is still applicable to the updated PRA. However, the revised PRA incorporated a revision to the internal flooding analysis with regard to human reliability analysis. The licensee stated that Byron has not yet implemented this procedure and has identified this as a regulatory commitment for Byron before implementation of the proposed CT, bypass test time, and STI changes.

Peer Review

The PWROG performed a peer review of the Braidwood PRA in September 1999 and completed a peer review of the Byron Station PRA in July 2000. The licensee stated that the PRA model for Byron and Braidwood is an integrated model and has been updated since completion of the peer reviews. The licensee implemented updates to the PRA in February 2002, December 2002, May 2003, June 2003, July 2005, and October 2007. The licensee updated the disposition of its remaining open “A” and “B” facts and observations (F&Os) in its RAI response. The licensee has either resolved the remaining F&Os or evaluated and found them to have no impact on the proposed CT, bypass test times, and STI changes.

PRA Update/Procedures

The licensee’s Risk Management Program, which includes Procedure ER-AA-600, “Risk Management,” and associated implementation procedures controls the PRA update process. Procedure ER-AA-600-1015, “FPIE PRA Model Update,” provides responsibilities and guidelines for updates to full-power internal events PRA models. This process allows for the evaluation of design, engineering, maintenance, and plant-specific risk parameters to be reviewed and the impact on the PRA to be assessed. PRA quality assurance provides guidance on documentation, storage, bases documents, work control, and updates. Based on the guidance, the licensee nominally schedules PRA model updates for approximately every 4 years, with the need for interim updates also controlled by procedure.

(II) PRA Results and Insights

Cumulative Risk

WCAP-15376 evaluated the cumulative CDF risk from the pre-TOP condition STIs and CTs to WCAP-15376 implementation. For this case, the cumulative impact on the CDF for 2-out-of-4 logic was within the RG 1.174 acceptance guidelines of less than1E-6/year, representing a very small change. The cumulative impact on CDF for 2-out-of-3 logic was slightly above the RG 1.174 acceptance guideline for a very small change, but within the acceptance guidelines for a small change. For Byron and Braidwood, the cumulative risk is limited from the TOP condition to WCAP-15376 implementation. Since the proposed change for Byron and Braidwood is from

TOP to WCAP-15376, the expected change in cumulative risk is expected to be less than the WCAP-15376 estimates.

External Events

The licensee evaluated the proposed RPS and ESFAS CTs, test bypass times, and STIs for their potential impact on external events, including fire; seismic events; and high winds, floods, and other (HFO) events for Byron and Braidwood. The following paragraphs discuss these events.

Fires

As stated by the licensee, internal fires were the dominant risk contributor in the IPEEE. The licensee performed the fire risk evaluation for the Byron and Braidwood using the Electric Power Research Institute fire-induced vulnerability evaluation (FIVE) method. The IPEEE lists the fire risk evaluation as follows:

- 5.0E-6/year for Byron Station Unit 1
- 6.1E-6/year for Byron Station Unit 2
- 4E-6/year for Braidwood Station Units 1 and 2

The proposed changes are associated with equipment and system unavailabilities associated with the proposed CTs and test bypass times and the failure of a remaining protection train, and the fire risk during the CT or test bypass time. Based on the estimated fire risk for Byron and Braidwood, the fire risk impact on the proposed CTs, bypass test times, and STI changes is small for the proposed TS changes.

Seismic Events

The licensee based the Byron and Braidwood seismic analysis on a seismic margins assessment (SMA) for the IPEEE submittal. Therefore, the licensee provided no quantitative estimate of the seismic contribution to plant CDF. The IPEEE seismic margin evaluation identified no vulnerabilities in the Byron and Braidwood IPEEE. To confirm that the total seismic risk at Byron and Braidwood is small, the NRC staff performed an independent simplistic calculation to estimate the magnitude of the seismic risk. The approach uses the plant's high confidence in low probability of failure (HCLPF) value determined by the licensee's SMA and the site's seismic hazard curve that is based on NUREG-1488, "Revised Livermore Seismic Hazard Estimate for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," issued April 1994, to derive an approximation of the magnitude of the risk associated with seismic events. The NRC staff's independent simplistic calculation, using an HCLPF value of 0.3g peak ground acceleration, estimated a seismic CDF of about 5E-6/year for Byron. The estimated seismic CDF is also bounding for Braidwood based on the higher NUREG-1488 values for Byron Station. Based on the above, the NRC staff expects that the impact of the proposed CT, bypass test times, and STI changes on seismic risk is small.

High Winds, Floods, and Other External Events

The IPEEE for Byron and Braidwood stated that HFO events were evaluated using the progressive screening approach described in NUREG-1407, "Procedure and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," issued June 1991, and GL 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," Supplement 4, dated November 23, 1988, to demonstrate that each plant meets the criteria in the 1981 edition of the SRP. In accordance with NUREG-1407, if a plant meets the SRP criteria, licensees can screen out HFO external events as a significant contributor to total CDF. Based on the IPEEE HFO evaluation results, the contribution to core damage from the increased CT, bypass test time, and STI changes is also expected to be small.

Total Risk Contribution

The NRC staff considered whether the estimated fire and seismic risk, in conjunction with the Byron and Braidwood internal event risk, could exceed the RG 1.174 base CDF of 1E-4/year with the implementation of WCAP-14333 and WCAP-15376. The estimated combined total CDF is estimated to be about 5.1E-5/year (4.0E-5/year + 6.1E-6/year + 5E-6/year) for Byron and about 3.7E-5/year (2.8E-5/year + 4E-6 + 5E-6) for Braidwood. RG 1.174 states that while there is no requirement to calculate the total CDF, if there is an indication that the CDF may be considerably higher than 1E-4/year, the focus should be on finding ways to decrease rather than increase risk. Given the typically conservative nature of the FIVE analysis methodology and the estimation of the seismic risk, the NRC staff finds that the total CDF is not expected to be considerably higher than 1E-4/year and is acceptable for this application. The NRC staff's conclusion assumes that the licensee implements the enhanced internal flooding response procedures before incorporating the proposed CT, bypass test time, and STI changes for Byron. The licensee identifies this as a regulatory commitment.

3.4.2.1.2 Tier 2 - Avoidance of Risk-Significant Plant Configurations

A licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is taken out of service in accordance with the proposed TS change.

Based on WCAP-14333, WCAP-15376, and licensee evaluations, including the functional units not evaluated generically by WCAP-14333, the licensee identified the following Tier 2 conditions as regulatory commitments:

- The licensee will implement administrative controls to ensure that activities that degrade the availability of the reactor coolant system (RCS) pressure relief system, auxiliary feedwater (AFW) system, ATWS mitigating system actuation circuitry (AMSAC), or turbine trip should not be scheduled when a logic train or an RTB train is inoperable for maintenance.

- The licensee will implement administrative controls to ensure that one complete emergency core cooling system (ECCS) train that can be actuated automatically must be maintained when a logic train is inoperable for maintenance.

- The licensee will implement administrative controls to ensure that activities that cause RTS and ESFAS master relays or slave relays in the available train to be unavailable, and activities that cause RTS and ESFAS analog channels to be unavailable, will not be scheduled when a logic train or an RTB train is inoperable for maintenance, with the exception of ESFAS Functions 2.c and 3.b.(3).
- The licensee will implement administrative controls to ensure that activities that result in the inoperability of electrical systems (e.g., alternating current and direct current power) and cooling systems (e.g., essential service water and component cooling water) that support the RCS pressure relief system, AFW system, AMSAC, turbine trip, one complete train of the ECCS, and the available reactor trip and ESFAS actuation functions

will not be scheduled when a logic train or an RTB train is inoperable for maintenance. That is, one complete train of a function that supports a complete train of a function noted above must be available.

- The licensee will implement enhanced internal flood response procedures at Byron.

- The licensee will trend and evaluate as-found and as-left data for three trip functions (i.e., OTDT, SG level, and pressurizer pressure) for 2 years (4 data points) following implementation of the proposed changes.

The licensee evaluated concurrent component outage configurations and confirmed the applicability of the Tier 2 restrictions for Byron and Braidwood. Based on the above, the NRC staff finds the licensee's Tier 2 analysis supports the implementation of WCAP-14333 and WCAP-15376 at Byron and Braidwood and satisfies the condition of the NRC staff's SERs for WCAP-14333 and WCAP-15376 regarding Tier 2.

3.4.2.1.3 Tier 3 - Risk-Informed Configuration Risk Management Program

Risk assessment of online configurations for both Byron and Braidwood uses the latest approved version of the Byron and Braidwood PRA model. Additionally, for signals not specifically modeled, the licensee is required by procedure to add the signal to the model or address the signal through surrogates.

Byron and Braidwood have implemented an online work control procedure that requires an integrated review to identify risk-significant plant configurations before and during maintenance activities. The licensee assesses risk based on the following evaluations:

- The licensee minimizes maintenance activities that affect redundant and diverse SSCs that provide backup for the same function.

- The licensee reviews the potential for planned activities to cause a plant transient and avoids work on SSCs required to mitigate the transient.

- The licensee does not schedule work if it is highly likely to exceed a TS CT requiring plant shutdown. For activities expected to exceed 50 percent of a TS allowed outage time, the licensee considers compensatory measures and contingency plans to minimize SSC unavailability and maximize SSC reliability.

- For Maintenance Rule high-risk-significant SSCs, the licensee evaluates the impact of the planned activity on the unavailability performance.

- A quantitative risk assessment is performed to ensure that the activity does not pose an unacceptable risk. The licensee classifies the risk results (both CDF and LERF) by color, based on the increased risk of the activity.

The licensee's online work control procedure is applicable to both planned maintenance activities and emergent conditions during plant operations. The licensee stated that the procedure considers equipment unavailability, operations including testing and load dispatch, and weather conditions.

The NRC staff finds that the licensee's program to control risk is capable of adequately assessing the activities being performed to ensure that high-risk plant configurations do not occur and/or compensatory actions are implemented if a high-risk plant configuration or condition should occur. As such, the licensee's program provides for the assessment and management of increased risk during maintenance activities as required by the Maintenance Rule (Section (a)(4)) and satisfies the RG 1.177 guidelines for a CRMP for the proposed change.

3.4.3 Implementation and Monitoring Program

RGs 1.174 and 1.177 also establish the need for an implementation and monitoring program to ensure that extensions to TS STI, CT, or bypass test times do not degrade operational safety over time and that no adverse effects occur from unanticipated degradation or common-cause mechanisms. The purpose of an implementation and monitoring program is to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of SSCs impacted by the change. In addition, the application of the three-tiered approach in evaluating the proposed CT and bypass test times provides additional assurance that the changes will not significantly impact the key principle of defense in depth.

The licensee monitors the reliability and availability of the RTS and ESFAS instrumentation under the Maintenance Rule (Section (a)(1)), which requires a licensee to monitor the performance or condition of SSCs against licensee-established goals. Based on the above, Byron and Braidwood satisfy the RG 1.174 and RG 1.177 guidelines for an implementation and monitoring program for the proposed change.

3.5 Comparison with Regulatory Guidance

The proposed changes conform to TSTF-411, Revision 1, and the analysis performed in WCAP-15376, as approved by the NRC staff, including limitations and conditions identified in the NRC staff SER. Additional proposed changes conform to TSTF-418, Revision 2, and the analysis performed in WCAP-14333, as approved by the NRC staff, including limitations and conditions identified in the NRC staff SER. As such, the implementation of WCAP-14333 and WCAP-15376 at Byron and Braidwood is within the RG 1.174 and RG 1.177 acceptance guidance for ΔCDF, ΔLERF, ICCDP, and ICLERP.

3.6 NRC Staff Findings and Conditions

The NRC staff finds that the licensee has demonstrated the applicability of WCAP-14333 and WCAP-15376 to Byron and Braidwood and has met the limitations and conditions as outlined in the NRC staff's SERs. The NRC staff found the risk impacts for ΔCDF, ΔLERF, ICCDP, and ICLERP as estimated by WCAP-14333 and WCAP-15376 to be applicable to Byron and Braidwood and within the acceptance guidelines for RG 1.174 and RG 1.177. The licensee showed the applicability of the specified functional units to the topical report evaluations and results. The licensee's Tier 2 analysis evaluated concurrent outage configurations and confirmed the applicability of the risk-significant configurations identified by the NRC staf's SER limitations and conditions and topical report analysis to ensure control of these configurations. The licensee's Tier 3 CRMP was found to be consistent with the RG 1.177 CRMP guidelines and the Maintenance Rule (Section (a)(4)) for the implementation of WCAP-14333 and WCAP-15376. The licensee monitors the reliability and availability of the RTS and ESFAS instrumentation under the Maintenance Rule (Section (a)(1)). Therefore, the staff finds the TS revisions proposed by the licensee are consistent with the CTs, bypass test times, and STIs approved for WCAP-14333 and WCAP-15376 and meet the staff SER conditions and limitations for WCAP-14333 and WCAP-15376.

4.0 REGULATORY COMMITMENTS

The licensee made the following regulatory commitments related to WCAP-14333 and WCAP-15376:

- The licensee will implement administrative controls to ensure that activities that degrade the availability of the RCS pressure relief system, AFW system, AMSAC, or turbine trip will not be scheduled when a logic train or an RTB train is inoperable for maintenance.

- The licensee will implement administrative controls to ensure that one complete ECCS train that can be actuated automatically must be maintained when a logic train is inoperable for maintenance.

- The licensee will implement administrative controls to ensure that activities that cause RTS and ESFAS master relays or slave relays in the available train to be unavailable, and activities that cause RTS and ESFAS analog channels to be unavailable, should not be scheduled when a logic train or an RTB train is inoperable for maintenance, with the exception of ESFAS Functions 2.c and 3.b.(3).

- The licensee will implement administrative controls to ensure that activities that result in the inoperability of electrical systems (e.g., alternating current and direct current power) and cooling systems (e.g., essential service water and component cooling water) that support the RCS pressure relief system, AFW system, AMSAC, turbine trip, one complete train of ECCS, and the available reactor trip and ESFAS actuation functions will not be scheduled when a logic train or an RTB train is inoperable for maintenance. That is, one complete train of a function that supports a complete train of a function noted above must be available.

- For Byron, the licensee will implement enhanced internal flood response procedures.

- The licensee will trend and evaluate as-found and as-left data for three trip functions (i.e., OTDT, SG level, and pressurizer pressure) for 2 years (4 data points) following implementation of the proposed changes.

The NRC staff finds that reasonable controls for the implementation and subsequent evaluation of the proposed changes pertaining to the above regulatory commitment(s) are best provided by the licensee's administrative processes, including its commitment management program. The above regulatory commitments do not warrant the creation of regulatory requirements (i.e., items requiring prior NRC approval of subsequent changes).

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

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 (72 FR 14305, March 31, 2007). 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 NRC staff finds that the TS revisions proposed by the licensee are consistent with the CTs, bypass test times, and surveillance intervals evaluated in WCAP-14333 and WCAP-15376 and meet the NRC staff's SER conditions for the implementation of WCAP-14333 and WCAP-15376. The staff finds that the topical report generic analyses are applicable to the licensee's plant. The estimates for ΔCDF, ΔLERF, ICCDP, and ICLERP were found to be applicable for Byron and Braidwood and within the acceptance guidance of RG 1.174 and RG 1.177. The Tier 2 conditions were found to be applicable to Byron and Braidwood and will be incorporated into plant programs and procedures consistent with WCAP-14333 and WCAP-15376 and the NRC staff's SER. Implementation of the licensee's Tier 3 CRMP is in accordance with the Maintenance Rule (Section (a)(4)) and is consistent with the CRMP guidance of RG 1.177. The implementation and monitoring program satisfies the RG 1.174 and RG 1.177 guidelines for for the proposed change. Therefore, based on the above evaluation, the NRC staff concludes that the proposed amendment to extend RPS and ESFAS CTs, bypass test times, and STIs is acceptable.

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 Contributors: C. Doutt, NRR
I. Ahmed, NRR

Date: January 29, 2008