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

May 27, 2010

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

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION CHANGES TO REFLECT CONTROL ROD DRIVE CONTROL SYSTEM UPGRADE (TAC NO. MD9762)

Dear Mr. Pardee:

The Commission has issued the enclosed Amendment No.273 to Renewed Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit 1 (TMI-1), in response to your application dated September 29, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML082800174), as supplemented by letters dated May 6, 2009 (ADAMS Accession No. ML091260765), June 23, 2009 (ADAMS Accession No. ML091750846), August 21, 2009 (ADAMS Accession No. ML092400175), September 17, 2009 (ADAMS Accession No. ML092600658), October 15, 2009 (ADAMS Accession No. ML092890470), and November 11, 2009 (ADAMS Accession No. ML093220864).

The proposed changes would revise the TMI-1 technical specifications (TSs) to reflect design changes resulting from the planned control rod drive control system digital upgrade project. In addition, the proposed amendment would revise the TS to remove all references to the axial power shaping rods to reflect changes resulting from their elimination from the TMI-1 reactor.

A copy of the related safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Peter Bamford

Peter J. Bamford, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures: 1. Amendment No.273 to DPR-50
2. Safety Evaluation

cc: Distribution via Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 273
License No. DPR-50

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee, formerly AmerGen Energy Company, LLC), dated September 29, 2008, supplemented by letters dated May 6, 2009, June 23, 2009, August 21, 2009, September 17, 2009, October 15, 2009, and November 11, 2009, 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 Renewed Facility Operating License No. DPR-50 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 273, are hereby incorporated in the license. The Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately and shall be implemented prior to exceeding cold shutdown following the fall 2011 (T1R19) refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License and Technical Specifications

Date of Issuance: May 27, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 273

RENEWED FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace the following page of the Renewed Facility Operating License with the revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

| Remove | Insert |
|---|---|
| Page 4 | Page 4 |

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

| Remove | Insert |
|---|---|
| 1-3 | 1-3 |
| 3-27a | 3-27a |
| 3-34 | 3-34 |
| 3-35 | 3-35 |
| 4-3 | 4-3 |
| 4-5 | 4-5 |
| 4-48 | 4-48 |
| 5-4 | 5-4 |

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 273are hereby incorporated in the license. The Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], submitted by letter dated May 17, 2006, is entitled: "Three Mile Island Nuclear Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 3." The set contains Safeguards Information protected under 10 CFR 73.21.

(4) Fire Protection

Exelon Generation Company shall implement and maintain in effect all provisions of the Fire Protection Program as described in the Updated FSAR for TMI-1.

Changes may be made to the Fire Protection Program without prior approval by the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Temporary changes to specific fire protection features which may be necessary to accomplish maintenance or modifications are acceptable provided that interim compensate measures are implemented.

(5) The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

a. Identification of a sampling schedule for the critical parameters and control points for these parameters;

b. Identification of the procedures used to measure the values of the critical parameters;

c. Identification of process sampling points;

d. Procedure for the recording and management of data;

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[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

### 1.4.2 REACTOR PROTECTION SYSTEM

The reactor protection system is **described in Section 7.1** of the Updated FSAR. It is that combination of protection channels and associated circuitry which forms the automatic system that protects the reactor by control rod trip. It includes the four protection channels, their associated instrument channel inputs, manual trip switch, all rod drive control protection trip breakers, and activating relays or coils.

### 1.4.3 PROTECTION CHANNEL

A PROTECTION CHANNEL as **described in Section 7.1** of the updated FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply units, amplifiers, and bistable modules provided for every reactor protection safety parameter) is a combination of instrument channels forming a single digital output to the protection system's coincidence logic. It includes a shutdown bypass circuit, a protection channel bypass circuit and a reactor trip module.

### 1.4.4 REACTOR PROTECTION SYSTEM LOGIC

This system utilizes reactor trip module relays (coils and contacts) in all four of the protection channels as **described in Section 7.1** of the updated FSAR, to provide reactor trip signals for de-energizing the four control rod drive trip breakers. The control rod drive trip breakers are arranged to provide a one-out-of-two-times-two logic. Each element of the one-out-of-two-times-two logic is controlled by a separate set of two-out-of-four logic contacts from the four reactor protection channels.

### 1.4.5 ENGINEERED SAFETY FEATURES SYSTEM

This system utilizes relay contact output from individual channels arranged in three analog sub-systems and two two-out-of-three logic sub-systems as shown in Figure 7.1-4 of the updated FSAR. The logic sub-system is wired to provide appropriate signals for the actuation of redundant engineered safety features equipment on a two-of-three basis for any given parameter.

### 1.4.6 DEGREE OF REDUNDANCY

The difference between the number of operable channels and the number of channels which, when tripped, will cause an automatic system trip.

## 1.5 <u>INSTRUMENTATION SURVEILLANCE</u>

### 1.5.1 TRIP TEST

A TRIP TEST is a test of logic elements in a protection channel to verify their associated trip action.

3.5.1.7.1 Power may be restored through the breaker with the failed trip feature for up to two hours for surveillance testing per T.S. 4.1.1.

3.5.1.8 Deleted

3.5.1.9 The reactor shall not be in the Startup mode or in a critical state unless both HSPS actuation logic trains associated with the Functional units listed in Table 3.5-1 are operable except as provided in Table 3.5-1,D.

3.5.1.9.1 With one HSPS actuation logic train inoperable, restore the train to OPERABLE or place the inoperable device in an actuated state within 72 hours or be in HOT SHUTDOWN within the next 12 hours. With both HSPS actuation logic trains inoperable, restore one train to OPERABLE within 1 hour or be in HOT SHUTDOWN within the next 6 hours.

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. The reactor trip, on loss of feedwater may be bypassed below 7% reactor power. The bypass is automatically removed when reactor power is raised above 7%. The reactor trip, on turbine trip, may be bypassed below 45% reactor power (Reference 1). The safety feature actuation system must have two analog channels functioning correctly prior to startup.

The anticipatory reactor trips on loss of feedwater pumps and turbine trip have been added to reduce the number of challenges to the safety valves and power operated relief valve but have not been credited in the safety analyses.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column "B" (Table 3.5-1). This is in agreement with redundancy and single failure criteria of IEEE 279 as described in FSAR Section 7.

There are four reactor protection channels. Normal trip logic is two out of four. Minimum required trip logic is one out of two.

f. If a control rod in the regulating group is declared inoperable per Specification 4.7.1.2, operation may continue provided that within 1 hour the rods in the group are positioned such that the rod that was declared inoperable is maintained within allowable group average position limits of Specification 4.7.1.2.

g. If the inoperable rod in Paragraph "e" above is in groups 5, 6, or 7, the other rods in the group may be trimmed to the same position. Normal operation of 100 percent of the thermal power allowable for the reactor coolant pump combination may then continue provided that within 1 hour the rod that was declared inoperable is maintained within allowable group average position limits in 3.5.2.5.

3.5.2.3 The worth of single inserted control rods during criticality is limited by the restriction of Specification 3.1.3.5 and the Control Rod Position Limits defined in Specification 3.5.2.5.

3.5.2.4 Quadrant Tilt:

a. Except for physics tests, the quadrant tilt, as determined using the full incore system (FIS), shall not exceed the values in the CORE OPERATING LIMITS REPORT.

The FIS is OPERABLE for monitoring quadrant tilt provided the number of valid symmetric string individual SPND signals in any one quadrant is not less than the limit in the CORE OPERATING LIMITS REPORT.

b. When the full incore system is not OPERABLE and except for physics tests quadrant tilt as determined using the power range channels for each quadrant (out of core detector system) (OCD), shall not exceed the values in CORE OPERATING LIMITS REPORT.

c. When neither detector system above is OPERABLE and, except for physics tests, quadrant tilt as determined using the minimum incore system (MIS), shall not exceed the values in the CORE OPERATING LIMITS REPORT.

d. Except for physics tests if quadrant tilt exceeds the tilt limit, allowable power shall be reduced 2 percent for each 1 percent tilt in excess of the tilt limit. For less than four pump operation, thermal power shall be reduced 2 percent below the thermal power allowable for the reactor coolant pump combination for each 1 percent tilt in excess of the tilt limit.

e. If quadrant power tilt exceeds the tilt limit then within a period of 10 hours, the quadrant power tilt shall be reduced to less than the tilt limit except for physics tests, or the following verifications and/or adjustments in setpoints and limits shall be made:

   1. Verify $F_Q$ (Z) and $F^N_{\Delta H}$ are within limits of the COLR once per 2 hours and restore QPT to ≤ steady state limit within 24 hours, or perform steps 2, 3, & 4 below.

3.5.2.5 Control Rod Positions

a. Operating rod group overlap shall not exceed 25 percent ±5 percent, between two sequential groups except for physics tests.

b. Position limits are specified for regulating control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified in the CORE OPERATING LIMITS REPORT.

1. If regulating rods are inserted in the restricted operating region, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within 24 hours, and FQ(Z) and $F^N_{\Delta H}$ shall be verified within limits once every 2 hours, or power shall be reduced to ≤ power allowed by insertion limits.

2. If regulating rods are .inserted in the unacceptable operating region, initiate boration within 15 minutes to restore SDM to ≥1% ΔK/K, and restore regulating rods to within restricted region within 2 hours or reduce power to ≤ power allowed by rod insertion limits.

c. Safety rod limits are given in 3.1.3.5.

3.5.2.6 Deleted

3.5.2.7 Axial Power Imbalance:

a. Except for physics tests the axial power imbalance, as determined using the full incore system (FIS), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

The FIS is operable for monitoring axial power imbalance provided the number of valid self powered neutron detector (SPND) signals in any one quadrant is not less than the limit in the CORE OPERATING LIMITS REPORT.

b. When the full incore detector system is not OPERABLE and except for physics tests axial power imbalance, as determined using the power range channels (out of core detector system)(OCD), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

c. When neither detector system above is OPERABLE and, except for physics tests axial power imbalance, as determined using the minimum incore system (MIS), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

d. Except for physics tests if axial power imbalance exceeds the envelope, corrective measures (reduction of imbalance by control rod movements and/or reduction in reactor power) shall be taken to maintain operation within the envelope. Verify FQ(Z) and $F^N_{\Delta H}$ are within limits of the COLR once per 2 hours when not within imbalance limits.

TABLE 4.1-1

INSTRUMENT SURVEILLANCE REQUIREMENTS

| CHANNEL DESCRIPTION | CHECK | TEST | CALIBRATE | REMARKS |
|---|---|---|---|---|
| 1. Protection Channel Coincidence Logic | NA | Q | NA | |
| 2. Control Rod Drive Trip Breaker | NA | Q | NA | (1) Includes independent testing of shunt trip and undervoltage trip features. |
| 3. Power Range Amplifier | D(1) | NA | (2) | (1) When reactor power is greater than 15%.<br>(2) When above 15% reactor power run a heat balance check once per shift. Heat balance calibration shall be performed whenever heat balance exceeds indicated neutron power by more than two percent. |
| 4. Power Range Channel | S | S/A | M(1)(2) | (1) When reactor power is greater than 60% verify imbalance using incore instrumentation.<br>(2) When above 15% reactor power calculate axial offset upper and lower chambers after each startup if not done within the previous seven days. |
| 5. Intermediate Range Channel | S(1) | P S/U | NA | (1) When in service. |
| 6. Source Range Channel | S(1) | P S/A | NA | (1) When in service. |
| 7. Reactor Coolant Temperature Channel | S | S/A | F | |

TABLE 4.1-1 (Continued)

| CHANNEL DESCRIPTION | CHECK | TEST | CALIBRATE | REMARKS |
|---|---|---|---|---|
| 19. Reactor Building Emergency Cooling and Isolation System Analog Channels | | | | |
| a. Reactor Building 4 psig Channels | S(1) | M(1) | F | (1) When CONTAINMENT INTEGRITY is required. |
| b. RCS Pressure 1600 psig | S(1) | M(1) | NA | (1) When RCS Pressure > 1800 psig. |
| c. Deleted | | | | |
| d. Reactor Bldg. 30 psi pressure switches | S(1) | M(1 | F | (1) When CONTAINMENT INTEGRITY is required. |
| e. Reactor Bldg. Purge Line High Radiation (AH-V-1A/D) | W(1) | M(1)(2) | F | (1) When CONTAINMENT INTEGRITY is required. |
| f. Line Break Isolation Signal (ICCW & NSCCW) | W(1) | M(1) | R | (1) When CONTAINMENT INTEGRITY is required. |
| 20. Reactor Building Spray System Logic Channel | NA | Q | NA | |
| 21. Reactor Building Spray 30 psig pressure switches | NA | M | F | |
| 22. Pressurizer Temperature Channels | S | NA | R | |
| 23. Control Rod Absolute Position | S(1) | NA | R | (1) Check with Relative Position Indication |
| a. Zone Reference Switch | NA | R(1) | NA | (1) Verify switch functions |
| 24. Control Rod Relative Position | S(1) | NA | NA | (1) Check with Absolute Position Indication |
| 25. Core Flooding Tanks | | | | |
| a. Pressure Channels Coolant | **NA** | NA | F | |
| b. Level Channels | **NA** | NA | F | |
| 26. Pressurizer Level Channels | S | NA | R | |

## 4.7 REACTOR CONTROL ROD SYSTEM TESTS

### 4.7.1 CONTROL ROD DRIVE SYSTEM FUNCTIONAL TESTS

Applicability

Applies to the surveillance of the control rod system.

Objective

To assure operability of the control rod system.

Specification

4.7.1.1 The control rod trip insertion time shall be measured for each control rod at either full flow or no flow conditions following each refueling outage prior to return to power. The maximum control rod trip insertion time for an operable control rod drive mechanism from the fully withdrawn position to ¾ insertion (104 inches travel) shall not exceed 1.66 seconds at hot reactor coolant full flow conditions or 1.40 seconds for the hot no flow conditions (Reference 1). If the trip insertion time above is not met, the rod shall be declared inoperable.

4.7.1.2 If a control rod is misaligned with its group average by more than an indicated nine inches, the rod shall be declared inoperable and the limits of Specification 3.5.2.2 shall apply. The rod with the greatest misalignment shall be evaluated first. The position of a rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments.

4.7.1.3 If a control rod cannot be exercised, or if it cannot be located with absolute or relative position indications, in or out limit indication, or zone reference switch indication, the rod shall be declared to be inoperable.

Bases

The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has actuated the 25% withdrawn reference switch during insertion from the fully withdrawn position. The specified trip time is based upon the safety analysis in UFSAR, Chapter 14 and the Accident Parameters as specified therein.

Each control rod drive mechanism shall be exercised by a movement of a minimum of 3% of travel every 92 days. This requirement shall apply to either a partial or fully withdrawn control rod at reactor operating conditions. Exercising the drive mechanisms in this manner provides assurance of reliability of the mechanisms.

## 5.3 REACTOR

Applicability

Applies to the design features of the reactor core and reactor coolant system.

Objective

To define the significant design features of the reactor core and reactor coolant system.

Specification

### 5.3.1 REACTOR CORE

5.3.1.1 A fuel assembly normally contains 208 fuel rods arranged in a 15 by 15 lattice. The reactor shall contain 177 fuel assemblies. Fuel rods shall be clad with zircaloy, ZIRLO, or zirconium-based M5 alloy materials and contain an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions. The details of the fuel assembly design are described in TMI-1 UFSAR Chapter 3.

5.3.1.2 The reactor core shall approximate a right circular cylinder with an equivalent diameter of 128.9 inches. The active fuel height is defined in TMI-1 UFSAR Chapter 3.

5.3.1.3 The core average and individual batch enrichments for the present cycle are described in TMI-1 UFSAR Chapter 3.

5.3.1.4 The control rod assemblies (CRA) are distributed in the reactor core as shown in TMI-1 FSAR Chapter 3. The CRA design data are also described in the UFSAR.

5.3.1.5 The TMI-1 core may contain burnable poison rod assemblies (BPRA) and gadolinia-urania integral burnable poison fuel pellets as described in TMI-1 UFSAR Chapter 3.

5.3.1.6 Reload fuel assemblies and rods shall conform to design and evaluation data described in the UFSAR. Enrichment shall not exceed a nominal 5.0 weight percent of $U_{235}$.

### 5.3.2 REACTOR COOLANT SYSTEM

5.3.2.1 The reactor coolant system shall be designed and constructed in accordance with code requirements. (Refer to UFSAR Chapter 4 for details of design and operation.)

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.273TO RENEWED

FACILITY OPERATING LICENSE NO. DPR-50

EXELON GENERATION COMPANY, LLC

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

DOCKET NO. 50-289

## 1.0 INTRODUCTION

By application dated September 29, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML082800174), as supplemented by letters dated May 6, 2009 (ADAMS Accession No. ML091260765), June 23, 2009 (ADAMS Accession No. ML091750846), August 21, 2009 (ADAMS Accession No. ML092400175), September 17, 2009 (ADAMS Accession No. ML092600658), October 15, 2009 (ADAMS Accession No. ML092890470), and November 11, 2009 (ADAMS Accession No. ML093220864), Exelon Generation Company, (Exelon, or the licensee)[1] requested changes to the technical specifications (TSs) for Three Mile Island Nuclear Station, Unit 1 (TMI-1). The supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 10, 2009 (74 FR 10308).

The proposed changes would revise the TMI-1 TSs to reflect design changes resulting from a planned control rod drive control system (CRDCS) digital upgrade project. In addition, the proposed amendment would revise the TS to remove all references to the axial power shaping rods (APSRs) to reflect changes resulting from their elimination from the TMI-1 reactor as part of the CRDCS upgrade. Specifically, the TS changes are as follows:

1) Modify TS 1.4.4 to provide reactor trip signals for de-energizing four breakers instead of six control rod drive reactor trip breakers (RTBs).

2) Delete the TS 3.5.1.8 and TS 3.5.1.8.1 requirements for silicon-controlled rectifier (SCR) electronic trips.

3) Delete the reference to the axial power shaping group in TS 3.5.2.2.f and the reference to Group 8 APSRs in TS 3.5.2.2.g.

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[1] The application dated September 29, 2008, was submitted by AmerGen Energy Company, LLC. Effective January 8, 2009, the license for TMI-1 was transferred from AmerGen Energy Company, LLC to Exelon Generation Company, LLC. By letter dated January 9, 2009, (ADAMS Accession No. ML090120538) Exelon Generation Company adopted and endorsed docketed submittals that requested specific licensing actions that were made by AmerGen.

4) Delete the TS 3.5.2.6 requirement to lock patch panels.

5) Modify TS 3.5.2.7.d to delete reference to APSRs.

6) Modify Table 4.1-1, CHANNEL DESCRIPTION No. 2, "Control Rod Drive Trip Breaker" to delete the reference to Regulating Rod Power SCRs.

7) Modify Table 4.1-1, CHANNEL DESCRIPTION No. 23, "Control Rod Absolute Position," and CHANNEL DESCRIPTION No. 24, "Control Rod Relative Positions," to add a new surveillance for the zone reference switches, delete the refueling calibration surveillance for Control Rod Relative Position, and change the word "Indicator" in the "Remarks" column to "Indication."

8) Modify TS 4.7.1.1 to delete the reference to APSRs.

9) Modify TS 4.7.1.3 to allow use of zone reference switches for locating a control rod.

10) Modify TS 5.3.1.4 to delete the references to APSRs.

## 2.0 REGULATORY EVALUATION

The construction permit for TMI-1 was issued by the Atomic Energy Commission (AEC) on May 18, 1968, and an operating license was issued on April 19, 1974. The plant design approval for the construction phase was based on the proposed General Design Criteria (GDC) published by the AEC in the *Federal Register* (32 FR 10213) on July 11, 1967 (hereinafter referred to as "draft GDC"). The AEC published the final rule that added Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "General Design Criteria for Nuclear Power Plants," in the *Federal Register* (36 FR 3255) on February 20, 1971 (hereinafter referred to as "final GDC" or just "GDC"). Differences between the draft GDC and final GDC included a consolidation from 70 to 64 criteria. In accordance with an NRC staff requirement memorandum from S. J. Chilk to J. M. Taylor, "SECY-92-223 - Resolution of Deviations Identified During the Systematic Evaluation Program," dated September 18, 1992 (ADAMS Accession No. ML003763736), the Commission decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971, which includes TMI-1. The TMI-1 Updated Final Safety Analysis Report (UFSAR), Section 1.4, provides an evaluation of the design bases of TMI-1 against the draft GDC.

The following were used to evaluate the application:

GDC-12, "Suppression of reactor power oscillations," states that "the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations, which can result in conditions exceeding specified acceptable fuel design limits, are not possible or can be reliably and readily detected and suppressed." Draft GDC, Criterion 7, contains similar requirements.

GDC-13, "Instrumentation and Control," requires that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges. Draft GDC, Criterion 12 contains similar requirements.

GDC-20, "Protective System Functions," requires the protection system to be designed: (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences; and (2) to sense accident conditions and to initiate the operation of systems and components important to safety. Draft GDC, Criterion 14 contains similar requirements.

Title 10 of the *Code of Federal Regulations* (10 CFR), Paragraph 50.55a(h)(2) requires that, for plants with construction permits issued prior to January 1, 1971, such as TMI-1, the design of protection systems must meet the original licensing bases, or may meet Institute of Electrical and Electronics Engineers IEEE-603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," including the correction sheet dated January 30, 1995.

Paragraph 50.65(a)(1) of 10 CFR states that "each holder of a license to operate a nuclear power plant...shall monitor the performance or condition of structures, systems, or components...in a manner sufficient to provide reasonable assurance that such structures, systems, and components...are capable of fulfilling their intended functions."

Appendix B to Part 50 of 10 CFR, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," provides, in part, the necessary quality assurance program requirements for the design, manufacture, construction, and operation of structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

Section 50.62 of 10 CFR, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants," requires, in part, that the ATWS mitigation system be composed of equipment that is diverse from the reactor trip system.

Regulatory Guide (RG) 1.53, revision 2, "Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems," states that conformance with the requirements of IEEE Std 379-2000, "Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems," provides methods for satisfying the NRC regulations with respect to the application of the single-failure criterion to the electrical power, instrumentation, and control portions of nuclear power plant safety systems. As specified in the licensee's submittal, dated September 29, 2008, the TMI-1 RTB design meets the requirements of this RG.

RG 1.75, revision 3, "Physical Independence of Electrical Systems," describes a method acceptable to the NRC staff for complying with the NRC regulations with respect to the physical independence requirements of the circuits and electric equipment that comprise or are associated with safety systems. As specified in the licensee's submittal, dated September 29, 2008, the TMI-1 RTB design meets the requirements of this RG.

Electric Power Research Institute (EPRI) Topical Report (TR)-106439, "Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications," describes an approach for evaluation and acceptance of commercial software-based equipment in nuclear safety systems.

NRC Safety Evaluation titled, "Review of EPRI Topical Report TR-106439, Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications," determined that EPRI TR-106439 contains an acceptable method to the NRC staff for dedicating commercial grade digital equipment for use in nuclear power plant safety applications.

## 3.0 TECHNICAL EVALUATION

### 3.1 Removal of APSRs

As stated in the application, dated September 29, 2008, the original design of the TMI-1 reactor included eight APSRs to allow operators to control the axial power profile in the core within limits specified in TS. APSRs are similar to standard control rod assemblies with three major differences: 1) the APSRs use a weaker neutron absorber material (Inconel); 2) the neutron absorber section in APSRs is part-length instead of full-length; and 3) the control rod drive mechanisms for APSRs do not insert on a reactor trip signal. The axial power imbalance limits formerly in the TMI-1 TS have been relocated to the Core Operating Limits Report (COLR).

In 1978 (Cycle 4), TMI-1 transitioned from rodded core designs to non-rodded, feed-and-bleed core designs. This decision was based on internal operating experience that showed the APSRs, if not used properly, could exacerbate axial imbalance swings rather than stabilize them during a power transient. Consequently, TMI-1 instituted administrative guidelines to keep APSRs in a stationary position throughout the cycle, including during power transients. Since that time, axial imbalance swings have been successfully maintained within core operating limits using regulating control rods along with planned water additions (either boration or dilutions) and the naturally damped characteristics of non-rodded core designs. According to the licensee, with only one minor exception, APSRs have not been used at TMI-1 for axial imbalance control since 1994 (Cycle 10). Since 1994, TMI-1 has successfully maintained axial imbalance operating limits using regulating control rod groups alone.

The original design of Babcock and Wilcox (B&W) reactors included APSRs to account for possible axial xenon instabilities which could occur for certain core designs and scenarios, and that these instabilities could be damped using APSRs. These conclusions were based on analyses documented in B&W Topical Report BAW-10010, Parts 1-3, "Stability Margin for Xenon Oscillations," dated August 1969, February 1970, and June 1971. In the application dated September 29, 2008, the licensee states that these analyses were performed based on core designs, operating philosophies (i.e., rodded core design) and using neutronic codes from the 1970 timeframe. Analyses summarized in the B&W-10010 report indicated that there was no axial instability for a non-rodded core.

According to the licensee, the ability to maintain acceptable power distributions, and to control any tendency towards axial oscillations without the need for APSRs, has been demonstrated at TMI-1 over the past 13-plus years of operation. In addition, an evaluation of axial xenon stability and transient imbalance control using regulating control rods was performed using AREVA's

NRC-approved NEMO nuclear design code. The evaluation demonstrated that, for power reductions and for return to full power transients, axial power oscillations are naturally damped and there is no axial xenon instability (i.e., diverging axial power oscillation).

In addition, the licensee stated in its September 29, 2008, submittal that in the early 2000 timeframe, multiple fuel rod defects occurred in B&W units (including TMI-1 in 2003) due to pellet-clad-interaction during end-of-cycle APSR withdrawal maneuvers. Therefore, starting with Cycle 16 in 2005, APSRs have been withdrawn from the core at TMI-1 at beginning-of-cycle and have been parked in this position for the entire cycle with COLR limits preventing insertion.

The design functions of APSRs as described in the TMI-1 UFSAR, Section 1.4.7 and Appendix 13A, are to: 1) maintain an acceptable power distribution in the core and control any tendency towards axial oscillations; and 2) where practicable, create core flux imbalance during the power imbalance detector correlation (PIDC) test such that measurements can be taken to obtain information regarding the correlation between incore and excore detectors. The ability to suppress power oscillations is described in GDC-12, as well as draft GDC, Criterion 7 and the PIDC test ensures that the excore detectors are properly calibrated with respect to incore instrumentation.

The licensee conducted a review of the plant safety analyses in UFSAR Chapter 14 to ensure that the APSRs are not credited in any of the events analyzed for TMI-1 and that the removal of the APSRs does not impact any of the results of those analyses. In addition, review of the AREVA reload methods showed that they require an evaluation of cycle-specific core parameters against the assumptions used in UFSAR safety analyses to demonstrate that the safety analyses remain bounding for the reload. Typically, these cycle-specific values are determined based on the nominal position of the APSRs. For the TMI-1 cores with APSRs removed, these parameters would be calculated in accordance with AREVA reload methods with no APSR poison in the core.

In addition, as stated in the licensee's application dated September 29, 2008, the AREVA reload methods require at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience departure from nucleate boiling (DNB) during normal operation or events of moderate frequency. For example, an increase in bypass flow due to the removal of the APSRs reduces the volume of coolant that would be available to transfer heat from fuel rods and could adversely affect DNB calculations. A review of the current analyses of record demonstrated that sufficient retained thermal margin is available to offset the small increase in bypass flow that is calculated with the APSRs eliminated and conservatively assuming there are no burnable poison rod assemblies in the core. The AREVA reload methodology also indicated that the elimination of APSRs would increase bypass flow and reduce hydraulic lift of the fuel; therefore, for fuel assembly hold down considerations, the current analysis of record would remain bounding with APSRs eliminated.

As stated in the licensee's application dated September 29, 2008, AREVA loss-of-coolant-accident (LOCA) methods require that peak cladding temperature (PCT) does not exceed 2200 degrees Fahrenheit (F) based on an approved evaluation model analysis that incorporates 10 CFR 50 Appendix K models. An increase in bypass flow reduces the volume of coolant that would be available to transfer heat from fuel rods and could adversely affect LOCA PCT calculations. A review of the current analyses of record demonstrated a conservatively bounding bypass of 7.5% (maximum value consistent with AREVA LOCA methodology) was assumed in both the Mark-B12 and Mark-B-HTP LOCA analyses. This value remains bounding for a core with no APSRs, whether the core is all Mark-B12 (maximum bypass flow of 6.78%), all Mark-B-HTP (maximum bypass flow of 7.32%), or mixed Mark-B12/Mark-B-HTP (maximum bypass flow between 6.78 and 7.32%). Therefore, there is no reduction in the current margin to the 2200 degree PCT limit with APSRs removed.

Analyses have shown that the core designs employed at TMI-1 are stable with respect to axial oscillations and that xenon oscillations initiated during power transients are naturally damped. Actual operating experience at TMI-1 bears out the analysis conclusions that the axial imbalance control can be maintained using coordinated movements of regulating Control Rod Group 7 (CRG-7) control rods using timed water additions. Therefore, there is adequate assurance that GDC-12, and draft GDC, Criterion 7, can be met without APSRs. Additionally, the APSRs have not been used for transient imbalance control at TMI-1 since 1994. The axial imbalance swings required for successful performance of the PIDC test were analyzed AREVA Document Identifier 32-9031517-001, "Feasibility Study for PIDC Test Without APSRs," dated December 19, 2006, and proven to be obtainable using regulating CRG-7 control rods. Therefore, the ability to perform the PIDC test has been demonstrated, both analytically and in the plant, without the use of APSRs.

Based on the discussion above, the NRC staff concludes that the current licensing basis will remain valid for a core configuration with no APSRs. The NRC staff agrees that safety analyses do not credit or account for APSRs and reload licensing analyses remain valid for the TMI-1 changes in core bypass flow resulting from removal of APSRs. Therefore, elimination of APSRs from the TMI-1 reactor is acceptable.

The NRC staff reviewed the requested TS changes to ensure consistency with the analysis summarized above. The pertinent TS associated with the APSRs are TS Section 3.5.2.2, 3.5.2.7, 4.7.1.1, 4.7.1.3 and 5.3.1.4, as described in Section 1.0 of this evaluation. The TSs are being revised to remove all references to the APSRs. The staff concurs that these changes are appropriate and acceptable.

### 3.2 Reactor Trip Breaker Changes

The licensee intends to replace the existing General Electric (GE) models AK-15 and AK-25 RTBs with the new Square D Masterpact NT Breakers. This portion of the modification, as detailed in the amendment request, would modify TS 1.4.4 to provide reactor trip signals for de-energizing four breakers instead of six control rod drive RTBs.

The NRC staff has reviewed the licensee's regulatory and technical analyses in support of its proposed license amendment, which are described in Attachment 1 to the application dated September 29, 2008. The existing reactor protection system (RPS) monitors parameters for safe operation and trips the reactor to protect the reactor core against fuel rod cladding damage. The RPS logic and the interface between the RPS and the CRDCS are described in TMI-1 UFSAR, Section 7.1.2.2. The system consists of four identical protection channels (A, B, C and D), each terminating in a trip relay within a reactor trip module. The logic in each reactor trip module controls one or more RTBs in the control rod drive power system. RTBs A and B control all the 3-phase primary power to the rod drives; RTBs C and D control the direct current (DC) power to rod groups 1 through 4. Electronic trip assemblies (ETAs) E and F control gating power to regulating groups 5-7 as well as to the APSRs (group 8). The APSRs receive a trip signal but do not physically insert on a reactor trip.

The existing GE RTBs have undervoltage (UV) and shunt trip (ST) devices to provide diverse tripping methods. The UV coil receives its power from the protection channel associated with each breaker and is de-energized to trip the reactor. The ST coil is energized by action of a voltage sensing relay, which operates when a trip is initiated via the RPS logic.
The licensee requested the proposed change (i.e., replacement of RTBs) in connection with the replacement of the existing CRDCS with a digital CRDCS. The licensee is replacing two existing alternating current (AC) GE breakers and four DC GE breakers with four new AC Square D Masterpact NT circuit breakers. In the proposed amendment, the RTBs are not altered functionally, but are replaced with an upgraded type of hardware and the configuration of the trip logic is modified to accommodate the new CRDCS.

According to the amendment request, the revised RTB configuration will result in RTBs A and C being placed in series in one parallel power path, and RTBs B and D in series in the other parallel path. The 3-phase primary power to the rod drives will be through these two parallel paths. In the revised configuration, RPS Channels A, B, C and D will trip corresponding CRD breakers A, B, C, and D. The one-out-of-two-times-two trip logic of the original system is maintained. The new RTB configuration of four breakers maintains the safety design basis of the CRD system in that upon receipt of a trip signal, manual or from RPS, the RTBs are tripped, thus removing power to the control rods leading to a reactor trip.

Each Square D RTB has diverse trip devices including an UV device that is de-energized to trip and a ST device that is energized to trip. The UV coil receives its power from the protection channel associated with each breaker and is de-energized to trip the reactor. The ST coil is energized by action of a voltage sensing relay, which operates when a trip is initiated via the RPS logic. According to the licensee, the RPS and RTBs are diverse from other trip systems such as Diverse Scram System (DSS) and ATWS System, and this diversity will be maintained. As with the existing system configuration, no single breaker failure will prevent a reactor trip when required, or cause a spurious reactor trip.

The licensee stated in the LAR that the new RTBs are fully qualified for the safety-related application. In addition, the licensee provided the following information for the proposed design and for the new RTBs:

1) The existing channelization, separation, and independence of the RTBs and associated cables will be maintained.

2) The new RTBs are seismically qualified, will be seismically installed and located in areas not subject to harsh environments.

3) The new RTBs are an updated version of a type of hardware used and operated successfully in this application. They replace the original RTBs which have become obsolete, have many problems and are increasingly difficult to maintain.

In response to the NRC staff's request for additional information (RAI), the licensee provided additional clarification to the proposed design via letter dated May 6, 2009.

- In RAI question 1, the staff asked the licensee to provide a comparison of the operating modes of the UV device in the RTBs with the new RTBs. The licensee confirmed that the UV device in the new RTBs has a direct acting mechanical trip that is actuated upon low or loss of control voltage. This is similar to the existing design and therefore acceptable. The licensee also identified that the new UV and shunt devices have a microcontroller that does not introduce a different failure mode when compared to the existing design. This microcontroller is considered a "digital" upgrade to the system and is addressed more specifically in Sections 3.6-3.8 of this evaluation.

- NRC Information Notice (IN) No. 88-38, "Failure of Undervoltage Trip Attachment on General Electric Circuit Breakers," documents low design margins of the torque available to trip GE AK breakers with the UV device. In RAI question 2, the NRC staff asked the licensee to provide information on torque available from the shunt and UV trip devices, and the torque required to trip the proposed Square D breakers. In its response, the licensee provided the following test results based on the data from a sample Square D breaker

| | |
|---|---|
| UV Trip Device | |
| Minimum force measured from UV | 2.1 pounds (lbs) force |
| Maximum force measured to trip | 1.3 lbs force |
| Percent margin | 62% |
| | |
| ST Device at 125 volts (V) DC | |
| Minimum force measured from device | 6.3 lbs force |
| Maximum force measured to trip | 1.2 lbs force |
| Percent margin | 425% |
| | |
| ST Device at 95 V DC | |
| Minimum force measured from device | 3.2 lbs force |
| Maximum force measured to trip | 1.2 lbs force |
| Percent margin | 167% |

  Based on this information, the licensee concluded that there is adequate margin in trip force available from the UV and ST devices to consistently trip the RTBs, and that preventive maintenance coupled with breaker performance trending will preclude the type of failures associated with the existing GE breakers. The staff agrees that there is adequate trip force margin available and finds that the maintenance and performance trending described will provide reasonable assurance that the margin specified will be maintained.

- In RAI question 3, the NRC staff asked the licensee to provide information on breaker operating times and preventive measures to ensure that lubrication hardening problems experienced with GE breakers would not be experienced on the new breakers. The licensee initially indicated that Mobil 28 grease was used as a lubricant. In a supplemental letter dated June 23, 2009 (ADAMS Accession No. ML091750846), the licensee informed the staff that an inconsistency about the use of Mobil 28 grease as lubricant was noted during subsequent validation of information provided in the LAR and RAI response. The actual lubricant used for the Square D Masterpact NT breaker operating mechanism is the Mobil Mobilith SHC-100. The licensee further stated that aging tests have validated proper performance of the breaker with Mobil Mobilith SHC-100 lubricant. The licensee intends to trend the operating time of the breaker with independent actuating paths through the ST and UV trip and verify that the trip time of the breaker through either path is not degraded over a period of time. The staff finds this acceptable.

- In response to RAI question 4, regarding the NRC staff concerns related to common mode failure of the Masterpact breakers due to the radio frequency interference (RFI) or electromagnetic interference (EMI), the licensee stated that subsequent to the LAR submittal, Nuclear Logistic Inc. (NLI, the licensee's contractor) identified that the UV trip device and the ST device on the proposed Square D Masterpact breakers contain a firmware microcontroller. NLI tested the UV and ST devices for susceptibility to EMI/RFI for service conditions per Electric Power Research Institute (EPRI) Topical Report (TR)-102323, Revision 3, "Guideline for Electromagnetic Interference Testing of Power Plant Equipment," and verified that microcontroller and the coil operation were not affected. The licensee also confirmed that the Micrologic trip system that is typically supplied with Square D breakers is not installed or used in the TMI RTBs. These RTBs are not used for fault current protection but are essentially electrically operated switches. Hence, there are no additional actions necessary to maintain control over the hardware, software and procedures used to test and calibrate the Micrologic protective systems. The staff finds this response acceptable.

- In response to RAI question 5, the licensee confirmed that the maximum DC system voltage during battery charging will not exceed the 137 VDC rating of the ST devices. The staff finds this response acceptable.

- In RAI question 6, the NRC staff asked the licensee to provide clarification on the methods used to separate safety and non-safety related portions of the RTB control power. In its response to this question, the licensee identified the physical separation for cables, wirings, terminations and isolation devices or relays in circuits that require interface between safety and non-safety related circuits and confirmed that the required separation will be maintained according to station engineering guidelines. The staff finds this response acceptable.

- In response to RAI question 7, the NRC staff asked the licensee to provide information on Failure Modes and Effects Analysis (FMEA) that was performed for the Square D Masterpact NT breaker and the modified trip logic system. In its response, the licensee stated that a FMEA for the new RTB configuration, including the UV trip device and ST devices, was performed. The FMEA concluded that no single failure of an RTB component would prevent completion of the reactor trip function. The staff reviewed the licensee's response and agreed that based on the information provided, no single failure of an RTB component would prevent completion of the reactor trip function.

TS Section 1.4.4, "REACTOR PROTECTION SYSTEM LOGIC," has the following statement: "[t]his system utilizes reactor trip module relays (coils and contacts) in all four of the protection channels as described in Section 7.1 of the updated FSAR, to provide reactor trip signals for de-energizing the six control rod drive trip breakers." The proposed configuration of the RTBs will have four breakers and TS 1.4.4 will be modified to state "....energizing the four control rod drive trip breakers." The staff finds these changes acceptable because the one-out-of-two-times-two reactor trip logic of the original system is maintained in the proposed configuration using the four new RTBs.

The information provided by the licensee confirms that the failure of any one RTB will not inhibit the reactor trip function of the RTBs. The ST and UV trip relays associated with the RTBs will function in a manner similar to the existing design with higher margins for torque required to perform the trip function. The electrical performance capabilities of the proposed RTBs are equal to or better than the existing RTBs. It is, therefore, reasonable to expect that the modified configuration will retain the level of reliability credited in the original design.

Based on the above licensee responses to staff questions and the details provided on qualification and electrical capabilities of the proposed Square D RTBs, the staff finds that there is reasonable assurance that the proposed breaker replacement will maintain the required safety function of the RTBs (note: the microcontroller in the UV and shunt devices is addressed more specifically in Sections 3.6-3.8 of this evaluation).

The evaluation described in this section, and later in Sections 3.6-3.8, supports the conclusion that the proposed design will maintain compliance with 10 CFR 50.65(a)(1) in that the proposed modification of the RTBs will not alter the function of the protection system or the ability to monitor performance of the RTBs. The licensee will continue to have a protection system that meets the original licensing basis, as required by 10 CFR 50.55a(h)(2). The proposed design conforms with RG 1.75, revision 2, and RG 1.53, revision 3, which are methods acceptable to the staff for single failure considerations and physical independence of circuits and electrical equipment. The proposed TS changes address the planned changes in system configuration. The proposed change to TS 1.4.4, as described in Section 1.0 of this evaluation, does not alter the intent of the TS requirements, is consistent with the technical evaluation presented above, and is, therefore, acceptable.

### 3.3 Electronic Trips Associated with DC RTBs and the Diverse Scram System (DSS)

The proposed modification deletes the SCR-based ETAs described in TS 3.5.1.8 and TS 3.5.1.8.1. These specifications are being deleted because the control rod power SCR electronic trips that are associated with the DC RTBs will be removed as part of the reconfiguration. Appropriate TS Actions for the new RTBs are covered elsewhere in TS 3.5.1; therefore, the removal of the requirements for the ETA trips reflects the design of the new system and is acceptable to the NRC staff.

Similarly, TS Table 4.1-1, Function 2, "Control Rod Drive Trip Breaker and Regulating Rod Power SCRs," would be modified to delete Regulating Rod Power SCRs. The Regulating Rod Power SCRs are connected to the electronic trip assemblies that are associated with the DC RTBs. Since the DC RTBs would be deleted, the electronic trip assemblies that are associated with the DC RTBs would no longer be needed and can be deleted from TS Table 4.1-1, Function 2. Appropriate surveillances remain for the newly configured RTBs in TS Table 4.1-1 and therefore, this change is acceptable to the NRC staff.

The DSS would be modified to trip the DCRDCS electronic trip instead of tripping the breakers that provide power to the control rod drive system. The primary design requirement for the DSS is that it be diverse from the RPS. The diversity of the DSS from the RPS is maintained by this modification because the RPS would trip the incoming power breakers (RTBs) and the DSS would disable the power to the single rod power supplies (SRPSs) downstream of the RTBs. The independence of the RPS and DSS would be maintained because no common components are utilized by both trip systems. This would maintain compliance with the requirements of 10 CFR 50.62 and is, therefore, acceptable.

3.4 Software Control of Assignment of Control Rod Groups

The existing CRDCS includes connector patch panels to align individual control rod drive mechanisms and their corresponding control rod position indications with appropriate power supplies. TS 3.5.2.6 requires the patch panels to be locked at all times with limited access authorized by the Plant Manager. The DCRDCS would utilize software to assign control rods to specific control rod groups. As a result, the connector patch panels would no longer be needed and are proposed to be deleted. To modify DCRDCS control rod group assignments, the system must be offline with the reactor shutdown. The DCRDCS software would be modified, recompiled, and downloaded into the DCRDCS memory with password control. The software application used to reconfigure the control rod group assignments would be contained on a laptop computer in a locked cabinet in a vital area. Control rod group assignment verification would be performed by moving each control rod to physically verify correct group/rod assignments using diverse instrumentation (e.g. relative or absolute rod position indicators) following any software modification and download while the reactor is offline/subcritical/during refueling outages. Any attempt to download modified software into the DCRDCS during reactor operation would cause a reactor trip prior to the revised software taking effect. In order for the revised software to take effect, the DCRDCS processor must be restarted, which would trip the reactor by de-energizing the SRPS. Changing the control rod group assignments would be a configuration change that would be controlled by the licensee's design change process. The NRC staff concludes that the design change process and software controls provide adequate assurance that the control rod groups would remain in conformance with the requirements of the COLR. Therefore, TS 3.5.2.6 can be deleted

3.5 Zone Reference Switches and Control Room Displays

The zone reference switches, which are located in the position indication tube assembly, formerly did not display in a location readily available to the operator. With the planned modification, the indication from the zone reference switches will be available to the operator on the flat panel position indication display or the plant process computer via the DCRDCS. Group average meters would be deleted and group average position would be displayed on the flat panel position indication display. Inverter backed in-limit light emitting diodes (LEDs) would be connected on the position indication panel to provide in-limit indication independent from the normal flat panel position indication display power supply. In the event of a loss of offsite power, the reactor would trip and analog position indications would be lost. The inverter backed in-limit LEDs would then provide indication that each rod has inserted.

TS 4.7.1 requires control rod drive system functional tests. TS 4.7.1.3 requires that a control rod be declared inoperable if the control rod cannot be exercised or if it cannot be located. TS 4.7.1.3 would be modified to allow the use of zone reference switch indication in addition to absolute or relative position indication, and in/out limit indication, for determining the location of a control rod. This change reflects the addition of the Zone Reference Switch Indication to a display in the control room that is readily available to the operator. Since the licensee has also proposed TS-required testing for these switches on a refueling outage frequency in TS Table

4.1-1, Function 23, there is adequate assurance of switch functionality and therefore, its use can be credited in TS 4.7.1.3.

TS Table 4.1.1, Function 23, "Control Rod Absolute Position," would be modified to replace the word "Indicator" with "Indication" in the Remarks column that currently states "[c]heck with Relative Position Indicator." The replacement of the word "Indicator" with "Indication" in the remarks reflects the use of the flat panel position indication display instead of the existing analog meters. This is an administrative change that reflects the new design and is therefore acceptable to the NRC staff.

TS Table 4.1.1, Function 24, "Control Rod Relative Position," would be modified to delete the refueling calibration surveillance and to replace the word "Indicator" with "Indication" in the Remarks column. The relative position indication is driven by a digitally-based counter and has no adjustable hardware. Therefore, there would not be anything to calibrate and the refueling calibration for this device can be deleted. Operability would continue to be ensured by the TS-required channel checks performed each shift. Hence, this change is acceptable to the NRC staff. The replacement of the word "Indicator" with "Indication" in the remarks reflects the use of the flat panel position indication display instead of the existing analog meters. This is an administrative change that reflects the new design and is therefore acceptable to the NRC staff.

### 3.6 RTB Microcontrollers

The licensee's September 29, 2008, submittal stated, "[t]he replacement of the RTBs, although included in the overall modification, is not a digital upgrade." However, the licensee's May 6, 2009, submittal informed the staff that subsequent to the September 29, 2008, submittal, Nuclear Logistics, Inc. (NLI) notified the licensee that the RTBs contain microcontrollers. In the May 6, 2009, submittal, the licensee identified the use of microcontrollers in the RTBs as a digital upgrade to a safety system.

The microcontroller is a simple two wire device that acts as an electrically operated switch. The microcontroller is in a sealed unit, has no communication connection or interface with other systems, has no physical access provisions, and cannot be reprogrammed. There is no digital communication between the microcontroller and other devices in the RTB. Each RTB includes a shunt trip microcontroller and UV microcontroller. The shunt trip and UV microcontrollers are actuated using external electrical contacts.

The UV microcontroller receives power from the RPS. During normal operation, the UV microcontroller is programmed such that when energized it retracts the plunger against spring pressure. Upon loss of power, the microcontroller spring returns the plunger to its extended position, mechanically tripping the RTB. The only source of power to the UV microcontroller is the signal from the RPS. Voltage to the UV microcontroller is removed when the RPS trips, which would cause the plunger to release and trip its associated RTB. A hardware FMEA by Schneider/Square D and functional testing by NLI determined that there is no credible hardware or software (including Common Cause Failure (CCF)) failure mode of the UV microcontroller that could keep the RTB energized. Therefore, a failure of the UV microcontroller would not prevent the RTB from performing its safety function.

As a backup to the UV microcontroller, the shunt trip microcontroller provides a diverse trip of the RTB. During normal operation, the shunt trip microcontroller is de-energized. Upon loss of

RPS power, the shunt trip microcontroller is energized by 120 vdc and the plunger is extended against spring pressure to mechanically trip the RTB. A software failure of the shunt trip microcontroller could prevent operation of the shunt trip, which would cause a loss of the backup RTB trip. However, as stated above, the UV microcontroller would still trip the RTB; therefore, no loss of RTB safety function would occur. Failure of the shunt trip microcontroller would have the same effect as a mechanical coil failure, loss of power, or a blown fuse, which is similar to the current RTB design failure modes. Failure of the diverse shunt trip microcontroller would not prevent the tripping of the RTB by the UV microcontroller.

The RTB design was reviewed for credible CCFs and no credible CCFs were identified that would prevent the RTBs from performing their safety function. The licensee concludes, and the NRC staff agrees, that there is no failure mode, including a CCF of the microcontrollers on the UV and shunt devices, that could prevent the trip of the RTB when called for by the RPS.

## 3.7 EPRI TR-106439

As previously discussed, the new Schneider/Square D Masterpact NT, Model 08 NA, RTBs use programmable (firmware) microcontrollers in the UV and shunt trip devices. The hardware and firmware of the RTBs are commercial grade equipment being dedicated for safety-related applications by NLI (under the control of NLI's Nuclear Quality Assurance Program), using EPRI TR-106439, dated October 1996, as the basis for qualification of this equipment. In a safety evaluation dated July 17, 1997 (ADAMS Accession No. ML092190664), the NRC staff determined that EPRI TR-106439 contains an acceptable method for dedicating commercial grade digital equipment for use in nuclear power plant safety.

EPRI TR-106439 describes an approach for evaluation and acceptance of commercial software-based equipment in nuclear safety systems. EPRI TR-106439 includes critical characteristics (CCs), which are those important design, material, and performance characteristics of a commercial grade item needed to verify that it will perform in an equivalent manner as safety related equipment. Once these CCs are verified, then there is reasonable assurance that the item will perform its intended safety function. Translation of design requirements into CCs for a commercial grade item is a key element in the dedication process. For convenience purposes, EPRI TR-106439 separates the CCs into three groups: Physical Characteristics, Performance Characteristics, and Dependability Characteristics.

Physical Characteristics include the physical characteristics of the hardware such as size, mounting, and other characteristics similar to those for mechanical and measurement devices. Performance Characteristics include the functionality required of the device and performance related to this functionality. They also include environmental requirements related to the needed performance of the device. Dependability Characteristics include attributes that typically cannot be verified through inspection and testing alone and are generally affected by the process used to produce the device.

EPRI TR-106439 includes four examples (in Sections 6.1, 6.2, 6.3, and 6.4) to illustrate how the guidance can be applied for commercial digital items of varying complexity and safety significance. The examples range from a relatively simple digital meter up through an Engineered Safety Features Actuation System upgrade that is both relatively complex and of high safety significance. Each example includes CCs, acceptance criteria, and methods of verification.

EPRI TR-106439 Example 6.1 is for a simple device in which simplicity and testability of the commercial device and its function in the plant, coupled with widespread successful operating history, provide adequate assurance without the need for a commercial grade survey or detailed review of the device's internal design and development process. Example 6.2 is for an existing indicator that is being replaced with a new microprocessor-based device. Example 6.3 is for a multipurpose, highly configurable device that is used to perform a specific set of functions, based on software configuration developed by the utility for the application. Example 6.4 is for a complex digital device with a high safety significance of the application lead to a significantly higher level of effort required to evaluate and dedicate the device as compared to the previous examples.

NLI determined that the RTB microcontroller is a simple device that acts as an electrically operated switch and that this best fits under EPRI TR-106439 Example 6.2 with two additional CCs from Example 6.3.

Example 6.2 Physical Characteristics include:

- Configuration (model number, software revision, dimensions, and mounting); and
- Interfaces (input signal, input impedance, power, bargraph and digital display, setpoint adjustment, and contact output).

Example 6.2 Performance Characteristics include:

- Functionality (accuracy, range, and response);
- Functionality for Contact Output (setpoint adjustability, hysteresis, and response time);
- Environmental Compatibility (EMI/RFI, seismic, environment, radiation); and
- Behavior Under Abnormal/Faulted Conditions (loss of signal, loss of power, signal over/ under range).

Example 6.2 Dependability Characteristics include:

- Built-in Quality (maintenance of a Quality Assurance (QA) program, operating history, feedback program, application of a QA program, documented product history);
- Failure Modes and Failure Management (operating history, failure analysis, challenge testing);
- Configuration Control;
- Problem Reporting; and
- Reliability (reliability calculations, operating history).

The two Example 6.3 CCs that were analyzed for the RTB microcontrollers are: (a) Human-machine Interface under Performance Characteristics; and (b) Built-in Quality (vendor follows a digital system/software development process) under Dependability Characteristics.

## 3.8 RTB Microcontroller Application of EPRI TR-106439

In the November 11, 2009, submittal and its attachments, the licensee provided detailed information on each of the applicable EPRI TR-106439 CCs. These attachments are:

1) QR-06910327-1, "Qualification Report for the Square D Reactor Trip Switchgear for Three Mile Island," dated May 2009 (ML093220865) (Qualification report).

2) FAT-Report-06910327-1, "Factory Acceptance Testing Report on Square-D PZ4 Rx Trip Switchgear," dated August 2009 (ML093220866) (FAT report).

3) VVR-042181-1-Coil, "Verification and Validation Report for Square D Masterpact Circuit Breaker (Coils Only)," Dated October 2009 (ML093220867) (V&V report).

4) QR-042181-5, "EMI/RFI Qualification Report for Masterpact Circuit Breaker Shunt Trip and Undervoltage Trip," dated May 2009 (ML093220868) (EMI report).

The NRC staff reviewed the commercial grade dedication information supplied by the licensee to ensure that the microcontrollers were properly qualified. This review is summarized below:

EPRI TR-106439 Example 6.2 Characteristics:

Physical Characteristic Configuration (model number and software revision): The model number and software revision were confirmed by the NLI audit of the Schneider facility and review of the product literature. The model number and software revision information is contained in Section 2.2.1 of the V&V report.

Physical Characteristic Configuration (dimensions and mounting): The dimensions and mounting were confirmed by 100% of the coils being dedication tested and supplied in the RTBs. The dimensions and mounting information is contained on pages 40-51 of the FAT report.

Physical Characteristic Interfaces: The applicable interfaces are the coil wires and the plunger actuation to hit the trip bar. The interfaces were confirmed as correct for the application via a configuration review of the Schneider specifications and 100% of the coils being dedication tested and supplied in the RTBs. The interface information is contained in Table 6.1 of the V&V report and pages 44-50 of the FAT report.

Performance Characteristic Functionality: The functionality for the application was confirmed as correct by 100% of the coils being dedication tested and supplied in the RTBs. The functionality information is contained on pages 44-50 of the FAT report.

Performance Characteristic Environmental Compatibility (EMI/RFI, seismic, mild environment, and radiation): The environmental characteristics were confirmed via project specific qualification in accordance with the TMI-1 specification by testing and analysis. The EMI/RFI information is contained in the EMI report and the seismic, mild environment, and radiation qualification information is contained in the Qualification report.

Performance Characteristic Behavior Under Abnormal/Faulted Conditions (loss of power and voltage range): Loss of power was confirmed by removal and application of power as part of the dedication test of the equipment supplied to TMI-1. The voltage range was confirmed by operation across the plant specific voltage range as part of the dedication test of the equipment supplied to TMI-1. The loss of power and voltage range information is contained on pages 50 and 56 of the FAT report.

Dependability Characteristic Built-in Quality (maintenance of a QA program): Vendor maintenance of a documented QA program is demonstrated by Schneider maintaining a documented QA program that controls the lifecycle of the hardware and software. This was verified by NLI audit of the Schneider facilities. The maintenance of a QA program information is contained in Sections 6.1 and 7.1 of the V&V Report.

Dependability Characteristic Built-in Quality (operating history): The operating history shows that the microcontroller software (firmware) has been stable over the recent operating history and no software-related failures have been reported. The microcontroller software for the UV and shunt trip coils were originally issued in 2002. There have been no software revisions of these microcontrollers. NLI has supplied approximately 240 Masterpact circuit breakers with these microcontrollers installed in nuclear power plants applications (both safety and non-safety related). The breakers contain one or more of the shunt trip and UV microcontroller devices. There have been no problems reported to NLI. The commercial supplied base is over 50,000 units with no firmware failures reported to Schneider/Square D. The operating history information is contained in Section 7.3 of the V&V report.

Dependability Characteristic Built-in Quality (feedback program): The operating history of the vendor having a strong program to record feedback from problems in the field was verified by audit of the Schneider program. The NLI program is in accordance with the NLI Nuclear QA program. The feedback program information is contained in Sections 8.1 and 8.2 of the V&V report.

Dependability Characteristic Built-in Quality (application of a QA program): Evidence that the QA program was applied in the production of the procured items was verified by audit of the Schneider facilities. The audit provided the following NLI conclusions:

1) The Schneider design process was performed under the controls of the Schneider quality system.

2) Software design, development, and verification activities were performed under the controls of Schneider Procedure PAEL-G01, Revision C, Group Schneider Software Quality Assurance.

3) Hardware and software requirements were documented in accordance with Schneider Procedure 07, Revision D, Group Schneider Requirements Definition.

4) Design requirements were verified in accordance with Schneider Procedure 13, Revision D, Group Schneider Validation of Technical or Design Requirements.

5) Design validation was performed in accordance with documented, controlled procedures.

NLI determined that the guidance provided in these documents was comprehensive, clearly presented, and of sufficient detail to provide unambiguous requirements. The application of QA program information is contained in Sections 6.1 and 7.1 of the V&V report and the November 11, 2009, submittal's response to Question 9.

Dependability Characteristic Built-in Quality (documented product history): The documentation of the product operating history was reviewed by NLI. The microcontroller software (firmware) for the UV and shunt trip coils were originally issued in 2002. There have been no software revisions of these microcontrollers. NLI has supplied approximately 240 Masterpact circuit breakers with these microcontrollers installed that have been installed in nuclear power plants (both safety and non-safety related). The breakers contain one or more of the shunt trip and UV microcontroller devices. There have been no problems reported to NLI. The commercial supplied base is over 50,000 units with no firmware failures reported to Schneider/Square D. The documented product history information is contained in Section 7.3 of the V&V report.

Dependability Characteristic Failure Modes and Failure Management (operating history): The product operating history was reviewed concerning failure modes and failure management. The microcontroller software (firmware) for the UV and shunt trip coils were originally issued in 2002. There have been no software revisions of these microcontrollers. NLI has supplied approximately 240 Masterpact circuit breakers with these microcontrollers installed that have been installed in nuclear power plants (both safety and non-safety related). The breakers contain one or more of the shunt trip and UV microcontroller devices. There have been no problems reported to NLI. The commercial supplied base is over 50,000 units with no firmware failures reported to Schneider/Square D. The operating history information is contained in Section 7.3 of the V&V report.

Dependability Characteristic Failure Modes and Failure Management (failure analysis): Failure analysis identifying failure modes from the system standpoint and assessment of their significance was performed for hardware failure management and software failure management CCF analysis. Because the microcontroller is considered a simple device NLI did not perform a software failure modes and effects analysis. NLI's rationale for this decision is:

1) The vendor used a highly controlled process to develop and test software and the software/hardware system.

2) The installed base of over 50,000 units with this microcontroller have reported no software related failures.

3) The software has not been revised since being released in 2002.

4) A software CCF as described in Section 3.6 above would not prevent the microcontrollers from performing the RTB safety function.

NLI concluded that the firmware is highly reliable based on the following:

1) A highly controlled process was used to develop and test the software and the software/hardware system.

2) A highly controlled process was used during production of the coils.

3) Schneider emulation and black-box testing sufficiently verified compliance with design requirements.

4) The operating history identifies a highly reliable design.

5) No software/firmware failures have been identified during NLI testing.

6) No microcontrollers have been returned to NLI with failures due to software/firmware failures.

The failure analysis information is contained in Section 5.0 and 6.3 of the V&V report.

Dependability Characteristic Failure Modes and Failure Management (challenge testing): Challenge testing designed to test for possible critical failure modes in normal operation was performed by NLI for loss of voltage, degraded voltage, and abnormal conditions and events. The challenge testing information is contained in Section 5.0 of the V&V report.

Dependability Characteristic Configuration Control: The configuration control characteristic was confirmed by audit of the Schneider configuration control program and NLI's configuration control program in accordance with the NLI Nuclear QA Program. The configuration control information is contained in Sections 8.1 and 8.2 of the V&V report.

Dependability Characteristic Problem Reporting: The problem reporting characteristic was confirmed by audit of Schneider's formal problem reporting program and the existence of NLI's formal problem reporting program. The problem reporting information is contained in Sections 8.1 and 8.2 of the V&V report.

Dependability Characteristic Reliability (reliability calculations): The reliability calculations were confirmed via review of the reliability calculations during the NLI audit of the Schneider facility, which showed that the hardware reliability calculations were performed per MIL-HDBK-217F, and the determination that the software (firmware) is highly reliable. The reliability calculation information is contained in Section 5.0 of the V&V report.

Dependability Characteristic Reliability (operating history): The product operating history was reviewed concerning reliability. The microcontroller software (firmware) for the UV and shunt trip coils were originally issued in 2002. There have been no software revisions of these microcontrollers. NLI has supplied approximately 240 Masterpact circuit breakers with these microcontrollers installed in nuclear power plants applications (both safety and non-safety related). The breakers contain one or more of the shunt trip and UV microcontroller devices. There have been no problems reported to NLI. The commercial supplied base is over 50,000 units with no firmware failures reported to Schneider/Square D. The operating history information is contained in Section 7.3 of the V&V report.

EPRI TR-106439 Example 6.3 Characteristics:

The two Example 6.3 CCs that were analyzed for the RTB microcontrollers are Human-machine Interface under Performance Characteristics and Built-in Quality (vendor follows a digital system/software development process) under Dependability Characteristics.

Performance Characteristic Human-machine Interface: There are no human-machine interfaces associated with the RTB microcontrollers.

Dependability Characteristic Built-in Quality (vendor follows a digital system/software development process): The digital system/software development process includes document design requirements including software requirements, validation test reporting, software quality assurance procedures, software quality reviews, software production controls, coding specifications, and acceptance test requirements. The digital system/software development process was verified by the NLI audit of the Schneider facility. The audit provided the following conclusions:

1) The Schneider design process was performed under the controls of the Schneider quality system.

2) Software design, development, and verification activities were performed under the controls of Schneider Procedure PAEL-G01, Revision C, Group Schneider Software Quality Assurance.

3) Hardware and software requirements were documented in accordance with Schneider Procedure 07, Revision D, Group Schneider Requirements Definition.

4) Design requirements were verified in accordance with Schneider Procedure 13, Revision D, Group Schneider Validation of Technical or Design Requirements.

5) Design validation was performed in accordance with documented, controlled procedures.

NLI determined that the guidance provided in these documents was comprehensive, clearly presented, and of sufficient detail to provide unambiguous requirements. Information that the vendor follows a digital system/software development process information is contained in Sections 6.1 and 7.1 of the V&V report and the November 11, 2009, submittal's response to Question 9.

On the basis of its review of the V&V report, Qualification report, EMI report, and FAT report, the NRC staff concludes that the RTB commercial grade dedication program was performed in accordance with the guidelines of NRC approved EPRI TR-106439 and that the CCs analyzed were appropriate for the complexity and safety significance of the RTBs. The licensee's commercial grade dedication of the Schneider/Square D Masterpact NT, Model 08 NA, RTBs is a suitable acceptance process that has demonstrated that the RTBs will perform their intended safety function, and they are deemed equivalent to items designed and manufactured under a 10 CFR Part 50 Appendix B quality assurance program.

## 4.0 STATE CONSULTATION

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

## 5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (74 FR 10308). Accordingly, the amendment meets 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 amendment.

## 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: A. Attard
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G. Matharu
P. Bamford

Date: May 27, 2010

May 27, 2010

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

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION CHANGES TO REFLECT CONTROL ROD DRIVE CONTROL SYSTEM UPGRADE (TAC NO. MD9762)

Dear Mr. Pardee:

The Commission has issued the enclosed Amendment No. 273 to Renewed Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit 1 (TMI-1), in response to your application dated September 29, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML082800174), as supplemented by letters dated May 6, 2009 (ADAMS Accession No. ML091260765), June 23, 2009 (ADAMS Accession No. ML091750846), August 21, 2009 (ADAMS Accession No. ML092400175), September 17, 2009 (ADAMS Accession No. ML092600658), October 15, 2009 (ADAMS Accession No. ML092890470), and November 11, 2009 (ADAMS Accession No. ML093220864).

The proposed changes would revise the TMI-1 technical specifications (TSs) to reflect design changes resulting from the planned control rod drive control system digital upgrade project. In addition, the proposed amendment would revise the TS to remove all references to the axial power shaping rods to reflect changes resulting from their elimination from the TMI-1 reactor.

A copy of the related safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
***/ra/***
Peter J. Bamford, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures: 1. Amendment No. 273 to DPR-50
2. Safety Evaluation

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ADAMS Accession No.: ML0 92740791 * concurrence via memo

| OFFICE | LPLI-2/PM | LPLI-2/LA | EEEB/BC | EICB/BC | SNPB/BC | ITSB/BC | OGC | LPL1-2/BC |
|---|---|---|---|---|---|---|---|---|
| NAME | PBamford | ABaxter | GWilson* | WKemper* | AMendiola* | RElliott | BMizuno | HChernoff |
| DATE | 5/10/10 | 5/13/10 | 8/31/09 | 4/30/10 | 8/10/09 | 5/24/10 | 5/17/10 | 5/27/10 |

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