UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

October 8, 2009

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

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2
EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT
05000373/2009006; 05000374/2009006(DRS)

Dear Mr. Pardee:

On September 17, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications Inspection at your LaSalle County Station. The enclosed report documents the inspection results, which were discussed on September 17, 2009, with Mr. D. Rhoades and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and to compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, one NRC-identified finding of very low safety-significance was identified. The finding involved a violation of NRC requirements. However, because of its very low safety-significance, and because the issue was entered into your corrective action program, the NRC is treating the issue as a Non-Cited Violation (NCV) in accordance with Section VI.A.1 of the NRC Enforcement Policy.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the LaSalle County Station. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the LaSalle County Station. The information that you provide will be considered in accordance with Inspection Manual Chapter 0305.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Robert Daley, Chief
Engineering Branch 3
Division of Reactor Safety

Docket Nos. 50-373; 50-374
License Nos. NPF-11; NPF-18

Enclosure: Inspection Report No. 05000373/2009006(DRS); 05000374/2009006 (DRS)
w/Attachment: Supplemental Information

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Enclosure

U.S. NUCLEAR REGULATORY COMMISSION
REGION III

Docket No:	05000373; 05000374
License No:	NPF-11; NPF-18
Report No:	05000373/2009006(DRS); 05000374/2009006(DRS)
Licensee:	Exelon Generation Company, LLC
Facility:	LaSalle County Station, Units 1 and 2
Location:	Marseilles, IL
Dates:	August 31, 2009 through September 17, 2009
Inspectors:	Alan Dahbur, Senior Reactor Inspector (Lead) Ron Langstaff, Senior Reactor Inspector Dariusz Szwarc, Reactor Inspector
Approved by:	R. Daley, Chief Engineering Branch 3 Division of Reactor Safety (DRS)

Enclosure

SUMMARY OF FINDINGS

IR 05000373/2009006 (DRS); 05000374/2009006 (DRS); 08/31/2009 – 09/17/2009; LaSalle County Station, Units 1 and 2; Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications.

This report covers a two-week announced baseline inspection on Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications. The inspection was conducted by Region III based engineering inspectors. Based on the results of this inspection, one finding of very low safety-significance (Green) was identified. The finding was considered a Non-Cited Violation (NCV) of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. A finding of very low safety-significance (Green) and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedure, and Drawings" was identified by the inspectors for the licensee's failure to ensure that specific requirements of the Technical Requirement Manual (TRM) Section 3.8.b. would be performed as written. Specifically, the licensee inappropriately revised the applicability section of TRM 3.8.b. and added that the TRM actions were applicable when the circuit breaker associated with an inoperable protective device was closed and receiving power from an energized source. This revision did not ensure that verification of circuit breaker position would be performed on a periodic basis when/if the breaker was open. The licensee subsequently entered the issue into their corrective action program as AR 00961522.

  The finding was determined to be more than minor because the finding, if left uncorrected, would become a more significant safety concern. The revised applicability section of TRM 3.8.b did not ensure that necessary action, i.e. verification of circuit breaker position, would be performed to ensure that primary containment electrical penetrations would not be subject to excessive fault currents. The finding was of very low safety-significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." This finding was not associated with a cross-cutting aspect because the finding was not indicative of the licensee's current performance. (Section 1R17.1.b(1))

B. Licensee-Identified Violations

No findings of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R17 Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications (71111.17)

.1 Evaluation of Changes, Tests, or Experiments

a. Inspection Scope

From August 31, 2009 through September 17, 2009, the inspectors reviewed three safety evaluations (SEs) performed pursuant to 10 CFR 50.59 to determine if the evaluations were adequate and that prior NRC approval was obtained as appropriate. The inspectors could not review the minimum sample size of six SEs, because the licensee only performed a total of three SEs during the triennial sample period that were not previously reviewed. The inspectors also reviewed 17 screenings and one calculation where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. The inspectors reviewed these documents to determine if:

- the changes, tests, or experiments performed were evaluated in accordance with 10 CFR 50.59 and that sufficient documentation existed to confirm that a license amendment was not required;
- the safety issue requiring the change, tests or experiment was resolved;
- the licensee conclusions for evaluations of changes, tests, or experiments were correct and consistent with 10 CFR 50.59; and
- the design and licensing basis documentation was updated to reflect the change.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, to determine acceptability of the completed evaluations, and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," dated November 2000. The inspectors also consulted Part 9900 of the NRC Inspection Manual, "10 CFR Guidance for 10 CFR 50.59, Changes, Tests, and Experiments."

This inspection constituted three samples of evaluations and 18 samples of changes as defined in IP 71111.17-04.

b. Findings

(1) Inadequate Revision for Technical Requirement Manual Section

Introduction: A finding of very low safety-significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings" was identified by the inspectors for the licensee's inadequate revision of Technical Requirement Manual

Section 3.8.b. Specifically, the licensee inappropriately revised the applicability section of TRM 3.8.b. and added that the TRM actions were applicable to inoperable primary containment penetration conductor over-current protective devices when its circuit breaker was closed and receiving power from an energized source. This revision did not ensure that verification of circuit breaker position associated with inoperable protective devices would be performed on a periodic basis as required per the TRM.

Description: The function of the primary containment is to isolate and contain fission products released from the Reactor Primary System following a design basis Loss of Coolant Accident (LOCA) and confine the postulated release of radioactive material within limits. The isolation devices for penetrations in the primary containment boundary are part of the primary containment leak tight barrier.

If an electrical fault were to occur in an electrical circuit penetrating primary containment, the fault current could result in an energy release high enough to cause weakening or failure of penetration seals. In order to protect the leak tight integrity of the primary containment electrical penetrations, at LaSalle County Station, circuits were either provided with over-current protection or maintained de-energized.

Technical Requirements Manual 3.8.b. required that the primary containment penetration conductor over-current primary and backup devices be OPERABLE. The TRM basis indicated that compliance with the TRM Limiting Condition for Operation (TLCO) helped ensure that the primary containment electrical penetrations were protected from potentially damaging over-current conditions so that they will be maintained structurally sound and will limit leakage.

The TRM 3.8.b. basis indicated that if it was discovered that one or more primary containment penetration conductor over-current protective devices were inoperable, the device(s) must be restored to an OPERABLE status within 72 hours (Required Action A.1). Alternately, the associated circuit must be de-energized by either tripping, or racking out and removing a circuit breaker in the affected circuit (Required Action A.2.1 or A.3.1). In the event that the affected circuit was de-energized in accordance with Required Action A.2.1 or A.3.1, the associated circuit breaker must be verified to be tripped, or racked out and removed, on a periodic basis. This was necessary to ensure that the primary containment electrical penetrations will not be subjected to excessive fault currents. The completion time of once per seven days was appropriate because the devices were operated under administrative controls and the probability of their misalignment was low.

The inspectors identified that previously, the TLCO for TRM 3.8.b was applicable in MODES 1, 2, and 3. In March of 2007, the licensee revised the applicability section for TRM 3.8.b. and added that the TLCO was applicable in MODES 1, 2, and 3, when the associated circuit breaker was closed and receiving power from an energized source. The licensee screened out this change per 50.59 Screening L07-71. The justification for this change was based on that it was not necessary to apply the TRM 3.8.b. specification to an open circuit breaker, because per the TRM basis document, de-energizing an affected circuit performed the intended function of the over-current protective device.

The inspectors were concerned that the revised applicability statement did not ensure that the TRM requirements will be met. Specifically, the requirement to verify the circuit breaker associated with an inoperable primary containment penetration conductor over-current protective device in the appropriate position (open or racked out) once every seven days

would not be performed when/if the breaker was open. The inspectors determined that bypassing this requirement would increase the probability that a breaker could be in the incorrect (closed) position, thereby increasing the possibility that the primary containment electrical penetrations could be subjected to excessive fault currents as stated in the TRM basis.

On September 1, 2009, the licensee entered this issue into their corrective action program as AR 00961522 "TRM 3.8.b TLCO Conflicts with Bases."

Analysis: The inspectors determined that the licensee's inadequate revision to the applicability section for TRM Section 3.8.b was contrary to the requirements of 10 CFR Part 50, Appendix B, Criterion V and was a performance deficiency. Specifically, the licensee did not ensure that the TRM requirement to verify that the breaker associated with the inoperable primary containment penetration conductor over-current protective device was open would be performed on a periodic basis, once per seven days.

The finding was determined to be more than minor because the finding, if left uncorrected, would become a more significant safety concern. Specifically, as stated in the TRM basis, the periodic verification of breaker position was necessary to ensure that primary containment electrical penetrations would not be subjected to excessive fault currents.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "significance Determination Process," Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," Tables 3b and 4a for the Mitigating Systems Cornerstone. The inspectors determined that the finding was of very low safety-significance (Green) because the finding did not involve a design or qualification deficiency, there was no actual loss of safety function, no single train loss of safety function for greater than the TS allowed outage time, and no risk due to external events.

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not indicative of the licensee's current performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures or drawings.

Contrary to the above, on September 1, 2009, the inspectors identified that the applicability section for TRM 3.8.b no longer ensured that the TRM requirement would be performed as written. Specifically, the inspectors identified that the licensee, on March 22, 2009, inappropriately revised the applicability section for TRM 3.8.b. such that the revised TRM no longer ensured that verification of the circuit breaker position associated with inoperable primary containment penetration conductor over-current protective devices would be verified on a periodic basis as required. Because this violation was of very low safety-significance and because it was entered into the licensee's corrective action program as AR 00945275, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC enforcement policy. (NCV 05000373/2009006-01; 05000374/2009006-01)

.2 Permanent Plant Modifications

a. Inspection Scope

From August 31, 2009 through September 17, 2009, the inspectors reviewed nine permanent plant modifications that had been installed in the plant during the last three years. This review included in-plant walkdowns for portions of the modified Emergency Diesel Generator (EDG) Fuel Oil Day Tank System; EDG Cooler Outlet Throttle valves; Emergency Pressurization Station for valves 2IN141/142; and Unit 1 Schram discharge volume vent and drum timing adjustment needle valve. The modifications were selected based upon risk significance, safety-significance, and complexity. The inspectors reviewed the modifications selected to determine if:

- the supporting design and licensing basis documentation was updated;
- the changes were in accordance with the specified design requirements;
- the procedures and training plans affected by the modification have been adequately updated;
- the test documentation as required by the applicable test programs has been updated; and
- post-modification testing adequately verified system operability and/or functionality.

The inspectors also used applicable industry standards to evaluate acceptability of the modifications. The list of modifications and other documents reviewed by the inspectors is included as an attachment to this report.

This inspection constituted nine permanent plant modification samples as defined in IP 71111.17-04.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

.1 Routine Review of Condition Reports

a. Inspection Scope

From August 31, 2009 through September 17, 2009, the inspectors reviewed Corrective Action Process documents that identified or were related to 10 CFR 50.59 evaluations and permanent plant modifications. The inspectors reviewed these documents to evaluate the effectiveness of corrective actions related to permanent pant modifications and evaluations for changes, tests, or experiments issues. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problems into the corrective action system. The specific corrective action documents that were sampled and reviewed by the inspectors are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4OA6 Meetings

.1 Exit Meeting Summary

On September 17, 2009, the inspectors presented the inspection results to Mr. D. Rhoades, and other members of the licensee staff. The licensee personnel acknowledged the inspection results presented and did not identify any proprietary content.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Rhoades, Plant Manager
J. Bashor, Site Engineering Director
S. Ballinger, Training
J. Gumnick, Radiation Protection
J. Houston, Regulatory Assurance
W. Hilton, Design Engineering Manager
K. Ihnen, Nuclear Oversight Manager
M. Peters, Design Engineering
V. Shah, Design Engineering Supervisor – Electrical
S. Shields, Regulatory Assurance
T. Simpkin, Regulatory Assurance Manager
K. Taber, Work Management Director
J. Vegara, Regulatory Assurance
H. Vinyard, Work Control Director
G. Wilhelmsen, Design Engineering
E. Zacharias, Design Engineering

Nuclear Regulatory Commission
G. Roach, Senior Resident Inspector
F. Ramirez, Resident Inspector
R. Daley, Chief, Engineering Branch 3, DRS

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000373/2009006–01; 05000374/2009006–01	NCV	Inadequate Revision for Technical Requirement Manual Section

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather, that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

10 CFR 50.59 EVALUATIONS

Number	Description or Title	Date or Revision
L08-105	Evaluation for GE Safety Communication SC06-01 Containment System Response (GEH 0000-0069-6598-R0) (LaSalle Design Analysis L-003352)	Revision 0
L08-168	Jumper VP Chillers Defective Chill Water Outlet Switches	Revision 0
L08-258	ATRIUM 10XM Lead Fuel Assemblies (EC 371970) and Nuclear Fuel (ATRIUM 10) Design Changes Beginning with L2C13	Revision 0

10 CFR 50.59 SCREENINGS

Number	Description or Title	Date or Revision
L07-034	Modifications to Valves 2IN141 & 2IN142 Plus Tie-Ins to Increase Nitrogen Supply at South & North Back-Up Bottle Banks for An ADS/LLS Interface Issue	Revision 1
L07-071	TRM 3.8.b	Revision 0
L07-111	DCR to Disposition Nonconformance of the “2B” Diesel Generator (DG) Heat Exchanger (2E22-S001) Floating End not Being Coated in Accordance with Drawing VPF 3411-080(1)	Revision 0
L07-127	Unit 1 Scram Discharge Volume Vent and Drain Timing Adjustment Needle Valve	Revision 0
L07-151	Loss of Service Water	Revision 5
L07-169	Reactor Water Cleanup (RWCU) Primary Containment Isolation (PCIS) Trip Setpoint Change	Revision 0
L07-248	Unit 1(2) Back-Up Scram Valve Test	Revision 2
L07-249	RCIC Lube Oil Cooler Operation with Maximum SBO Suppression Pool Temperature	Revision 0
L07-300	Revise NPSH Calculations L-001355 and L-002255 NPSH to include 1.5 ft. UHS drawdown due to a worst case 30-day evaporative period	Revision 0
L08-013	Filling, Venting, and Draining of Reactor Core Isolation Cooling System	Revision 0
L08-028	Add Vents to the RHR Service Water System	Revision 1
L08-123	Battery Charger Startup and Shutdown	Revision 0

10 CFR 50.59 SCREENINGS

Number	Description or Title	Date or Revision
L08-153	Revise Design Analysis CE-LS-003 and L-000333 for Valve 1(2)E22-F004	Revision 0
L08-248	UFSAR Change to Load Shedding for SBO	Revision 0
L09-015	Replacing Nitrogen bottles on Instrument Nitrogen System	Revision 0
L09-045	LGA-010 Failure to SCRAM	Revision 0
L09-054	Filling RCIC Downstream of 2E51-F013 Valve	Revision 0

MODIFICATIONS

Number	Description or Title	Date or Revision
EC 350960	Scram Discharge volume Vent & Drain Pilot Valve	Revision 3
EC 351367	Replace 250 VDC Breakers 1DC05E-3C and 2DC05E-3C	Revision 0
EC 351826	Add Vents to the RHR Service Water System 2E12-F305A and 2312-F306A	Revision 2
EC 357579	Addition of Maintenance Valve in Drywell for Unit 1 in System (New Valve to be Used in Lieu of 1IN129)	Revision 0
EC 361234	Breaker Setting Revision for Safety Related MOV 1E12-F009, 1E12-F099A/B and 2E12-F009, 2E12-F099A/B	Revision 0
EC 362630	Unit 1 Scram Discharge Volume Vent and Drain Timing Adjustment Needle Valve	Revision 0
EC 364648	Replace 0 DG Cooler Outlet Throttle Valve EPN 0DG006 With an Equivalent Valve	Revision 1
EC 364700	Design Analyses to Support Resolution of LLS/ADS Interface Issue	Revision 0
EC 366534	Revise Setpoint of RWCU Heat Exchanger Room Differential Temperature Isolation	Revision 0

CALCULATIONS

Number	Description or Title	Date or Revision
L-001355	LaSalle County Station CSCS Hydraulic Model	Revision 4H
NED-I-EIC-0158	Determination of Optimal Reactor Core Isolation Cooling Keep Filled Alarm Setpoint and Associated Technical Specification Allowable Value	Revision 1
NED-I-EIC-0198	HPCS, LPCS and LPCI Discharge Min Flow Bypass Differential Pressure Switch Setpoint Error Analysis	Revision 3

CORRECTIVE ACTION PROGRAM DOCUMENTS ISSUED DURING INSPECTION

Number	Description or Title	Date or Revision
AR 00861846	LOP-DG-03 50.59 Screening Missing from Record	September 04, 2009
AR 00960982	Safety Screening L07-248 Requires Clarification	September 02, 2009
AR 00961522	TRM 3.8.B TLCO Conflicts with Bases	September 01, 2009
AR 00965772	Minor Discrepancies in Calculation NED-I-EIC-0198	September 16, 2009

CORRECTIVE ACTION PROGRAM DOCUMENTS REVIEWED

Number	Description or Title	Date or Revision
AR 00238238	1TO04P – Breaker Indicates Tripped when it is Closed	July 22, 2004
AR 00465313	New Model Scram Dump Valves Required Timing Valve Adjustment	March 12, 2006
AR 00499484	1C11-F379 – SDV Upstrem Vent and Drain Pilot Valve	June 13, 2006
AR 00511232	Published Interrupting Rating of Installed MCCB	July 20, 2006
AR 00601924	The NRC Identified Potential Finding with 50.59 Eval	March 10, 2007
AR 00601954	Scram Air Header Depressurization Time Needs Re-Evaluated	March 10, 2007
AR 00630337	Procedure MA-AB-756-600 Inadequate for Drywell Head Bolts	May 16, 2007
AR 00772730	Document Update Required for Station Blackout	May 07, 2008
AR 00935468	0DG01A Flow Initially Low, But Operable	June 25, 2009
AR 00937425	Pre-NRC MOD/50.59 Check-IN Deficiency, LOA-EH-101 Screening	June 30, 2009

PROCEDURES

Number	Description or Title	Date or Revision
LES-GM-109	Inspection of 480 V Klockuer-Moller Motor Control Center	Revision 35
LOA-WS-101	Loss of Service Water	Revision 5
LOP-DC-01	Battery Charger Startup and Shutdown	Revision 33
LOP-IN-05	Replacing Nitrogen bottles on Instrument Nitrogen System	Revision 20
LOS-DG-R0	0 Diesel Generator 0DG01K, Twenty-Four Hour Run Surveillance	Revision 15
LOS-RP-SR1	Unit 1(2) Back-Up Scram Valve Test	Revision 2

WORK ORDERS

Number	Description or Title	Date or Revision
01053109	RWCU HX RMS Hi Diff Temp Outbrd Isol Ch	March 25, 2008
00956876	OP-PMT-Perform LOD-RD-R2	May 28, 2008

OTHER DOCUMENTS/REFERENCES

Number	Description or Title	Date or Revision
AT 848189-02	Check-in Self-Assessment Report Template	June 30, 2009

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
AR	Action Request
CFR	Code of Federal Regulations
CNO	Chief Nuclear Officer
DRS	Division of Reactor Safety
EC	Engineering Change
EDG	Emergency Diesel Generator
IMC	Inspection Manual Chapter
IR	Inspection Report
LOCA	Loss of Coolant Accident
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
PARS	Public Available Records System
SDP	Significance Determination Process
SE	Safety Evaluation
TLCO	TRM Limiting Condition for Operation
TRM	Technical Requirement Manual

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Robert Daley, Chief
Engineering Branch 3
Division of Reactor Safety

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