March 27, 2007

Mr. J. Conway
Site Vice President
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000263/2007006(DRS)

Dear Mr. Conway:

On March 1, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed a combined baseline inspection of the Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications at the Monticello Nuclear Generating Plant. The enclosed report documents the results of the inspection, which were discussed with Mr. J. Grubb, and others of your staff at the completion of the inspection on March 1, 2007.

The inspectors examined activities conducted under your license as they relate to safety and 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 the inspection, one NRC identified finding, which involved a violation of NRC requirements of very low safety significance, was identified. Because of the very low safety significance of the violation and the fact that the issue was entered into the licensee's corrective action program, the NRC is treating the finding as a Non-Cited Violation (NCV) in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

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

Sincerely,

/RA/

David E. Hills, Chief
Engineering Branch 1
Division of Reactor Safety

Docket No. 50-263
License No. DPR-22

Enclosure: Inspection Report 05000263/2007006(DRS)
w/Attachment: Supplemental Information

cc w/encl: M. Sellman, President and Chief Executive Officer
Manager, Nuclear Safety Assessment
J. Rogoff, Vice President, Counsel, and Secretary
Nuclear Asset Manager, Xcel Energy, Inc.
State Liaison Officer, Minnesota Department of Health
R. Nelson, President
Minnesota Environmental Control Citizens
Association (MECCA)
Commissioner, Minnesota Pollution Control Agency
D. Gruber, Auditor/Treasurer,
Wright County Government Center
Commissioner, Minnesota Department of Commerce
Manager - Environmental Protection Division
Minnesota Attorney General's Office

document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-263
License No:	DPR-22
Report No:	05000263/2007006(DRS)
Licensee:	Nuclear Management Company, LLC
Facility:	Monticello Nuclear Generating Plant
Location:	Monticello, Minnesota
Dates:	February 12, 2007 through March 1, 2007
Inspectors:	A. Dunlop, Senior Reactor Inspector T. Bilik, Reactor Inspector
Observers:	V. Meghani, Reactor Inspector
Approved by:	D. Hills, Chief Engineering Branch 1 Division of Reactor Safety (DRS)

SUMMARY OF FINDINGS

IR 05000263/2007006(DRS); 02/12/2007 through 03/01/2007; Monticello Nuclear Generating Plant. Evaluations of Changes, Tests, Experiments and Permanent plant modifications.

The inspection covered a 2-week announced baseline inspection on evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by two regional based engineering inspectors. One Green finding associated with a Non-Cited Violation (NCV) was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 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 3; dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

Green. The inspectors identified a Severity Level IV NCV for an inadequate 10 CFR 50.59, "Changes, Tests, and Experiments," evaluation resulting in failure to receive prior NRC approval for changes in licensed activities associated with protection of the emergency diesel generator exhaust stacks against tornado generated missiles. Specifically, the licensee did not provide an adequate response to the question posed in 10 CFR 50.59(c)(2)(viii), and did not demonstrate that the proposed change did not result in a departure from a method of evaluation described in the Final Safety Analysis Report (as updated) used in establishing the design bases or in the safety analyses. As part of the corrective actions, the licensee verified that the emergency diesel generators remained operable and initiated actions to submit a licensee amendment request for use of the new methodology.

Because the Significance Determination Process is not designed to assess the significance of violations that potentially impact or impede the regulatory process, this issue was dispositioned using the traditional enforcement process in accordance with Section IV of the NRC Enforcement Policy. However, the results of the violation, that is, the failure to demonstrate that the proposed change did not result in a departure from a method of evaluation, were assessed using the Significance Determination Process.

The finding was determined to be greater than minor because the change had the potential for impacting the NRC's ability to perform its regulatory function as the inspectors determined the change would have required prior NRC approval. The finding was of very low safety significance based on the completed analysis for the emergency diesel generator exhausts. This was determined to be a Severity Level IV NCV of 10 CFR 50.59. (Section 1R02)

B. Licensee-Identified Violations

No findings of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

.1 Review of 10 CFR 50.59 Evaluations and Screenings

a. Inspection Scope

From February 12, 2007, through March 1, 2007, the inspectors reviewed two evaluations performed pursuant to 10 CFR 50.59. The inspectors reviewed the evaluations to confirm that they were thorough and that prior NRC approval was obtained as appropriate. The inspector could not review the minimum sample size of five evaluations because the licensee only performed one evaluation during the biennial sample period. One additional safety evaluation was reviewed that was performed in the previous sample period for a total of two samples. The inspectors also reviewed 18 screenings where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. In addition, seven applicability determinations were reviewed to verify they did not meet the applicability requirements for a screening. The evaluations and screenings were chosen based on risk significance, safety significance, and complexity. The list of documents reviewed by the inspectors are included as an attachment to this report.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," and 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."

b. Findings

Inadequate 10 CFR 50.59 Evaluation for Diesel Generator Exhaust Missile Protection

Introduction: The inspectors identified an inadequate evaluation performed pursuant to 10 CFR 50.59 associated with the vulnerability of the emergency diesel generator (EDG) exhaust stacks to tornado generated missiles. Specifically, the licensee did not provide an adequate response to the question posed in 10 CFR 50.59(c)(2)(viii) and did not demonstrate that the proposed change did not result in a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analyses. This issue was considered to be of very low safety significance (Green) and was dispositioned as a Severity Level IV Non-Cited Violation (NCV).

Description: The inspectors reviewed 10 CFR 50.59 safety evaluation (SE) 03-004, concerning the utilization of the “TORMIS” probabilistic risk assessment (PRA) methodology (per Electric Power Research Institute (EPRI) Report NP-2005, Volumes 1 and 2). This methodology was to verify that the risk from tornado generated missiles was sufficiently small to justify leaving the EDG exhaust unprotected. On page 7 of SE 03-004 in Section III.8, the licensee responded to the question posed in 10 CFR 50.59(c)(2)(viii). This question asked, "Does the proposed change result in a departure from a method of evaluation described in the Final Safety Analysis Report (as updated) used in establishing the design bases or in the safety analyses”? The licensee justified the “No” answer to this question by citing the NRC acceptance of the EPRI methodology for specific plant features and subject to resolution of specific concerns in the NRC’s safety evaluation for EPRI Report NP-2005, dated October 26, 1983. The licensee’s evaluation included addressing the specific concerns and stated that the resolutions of these concerns for the Monticello plant were consistent with those accepted by the NRC for the D. C. Cook Nuclear Plant (Amendment No. 247 to DPR-58 and Amendment No. 228 to DPR-74).

The NRC’s safety evaluation concluded that the PRA methodology as contained in the EPRI report was an acceptable probabilistic approach for demonstrating compliance with the requirements of General Design Criteria 2 and 3 regarding protection of safety-related plant features from the effects of tornado and high wind generated missiles, but subject to the additional concerns identified. It further stated that use of the EPRI or any tornado missile probabilistic study should be limited to the evaluation of specific plant feature where additional costly tornado missile protective barriers or alternative systems were under consideration. The inspectors contacted the staff in the Office of Nuclear Reactor Regulation (NRR) to determine the basis for the NRC’s safety evaluation and the acceptability of the licensee using this methodology that was not in accordance with the current licensing basis. Based on this discussion, although the methodology had been reviewed and could be used as a basis for not having to physically protect specific plant features from tornado generated missiles, it was considered a change to the plant’s current licensing basis, which required a license amendment.

Based on the above, the inspectors concluded that the licensee use of NRC’s safety evaluation on the EPRI methodology was incorrect and that the licensee’s “No” answer to 10 CFR 50.59(c)(2)(viii), and the conclusion that “no activity requiring prior NRC approval per 10 CFR 50.59 was identified” were not justified.

The inspectors also determined that the results of the calculations based on the EPRI methodology discussed above were utilized for responses to the questions for 10 CFR 50.59(c)(2) (i) through (vi) in Section III of the SE 03-004 and that a USAR change was implemented to incorporate the use of TORMIS methodology. This finding also affected the licensee’s 10 CFR 50.59 screening SCR-04-0069, Revision 0, which was used to screen out activities involving subsequent application of the EPRI methodology for evaluation of other plant specific features from tornado generated missiles.

In response to the finding, the licensee initiated Action Request (AR) 01079705. The licensee determined that the NRC's 1983 safety evaluation endorsing the EPRI TORMIS methodology was misinterpreted by the licensee as a generic NRC approval for use and was inappropriately used in the 50.59 evaluation to conclude that prior NRC approval was not required. The licensee determined the EDGs remained operable based on the existing completed analysis and acceptance of similar technical approach by the NRC for other operating plants. The inspectors concluded that the licensee's determination was acceptable as the existing analysis using the TORMIS methodology did appear to address the limitations noted in the NRC's safety evaluation. The AR also recommended an action to submit an license amendment request to the NRC to incorporate the TORMIS methodology into the license basis for all the affected plant specific features.

Analysis: This issue was determined to involve a performance deficiency because the licensee incorrectly concluded that the TORMIS methodology had been approved for generic application and consequently concluded that prior NRC approval was not required when such a conclusion could not be supported by the documented 50.59 evaluation. Because violations of 10 CFR 50.59 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the significance determination process (SDP) described in Inspection Manual Chapter (IMC) 0609, "Significance Determination Process." The finding was determined to be greater than minor because the change had the potential for impacting the NRC's ability to perform its regulatory function as the inspectors determined the change would have required prior NRC approval.

The inspectors evaluated the finding using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, and determined that the finding screened as Green because it was not a design issue resulting in loss of function per Part 9900, Technical Guidance, "Operability Determinations, and Functionality Assessments for Resolution of Degraded, or Nonconforming Conditions Adverse to Quality or Safety," did not represent an actual loss of a system safety function, did not result in exceeding a technical specification allowed outage time, and did not affect external event mitigation. This was based on the licensee's operability determination that concluded that their use of the TORMIS methodology appeared to be consistent with the guidance provided in the NRC's safety evaluation of the methodology and that NRC had accepted its' use at other plants when used for the intended purpose. The inspectors did not identify a cross-cutting aspect with this finding.

Enforcement: Title 10 CFR 50.59(c)(2)(viii) states, in part, that a licensee shall obtain a license amendment pursuant to Section 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would result in a departure from a method of evaluation described in the Final Safety Analysis Report (as updated) used in establishing the design bases or in the safety analyses.

Contrary to the above, on July 28, 2003, the licensee approved a 10 CFR 50.59 evaluation (SE-03-004) incorporating a change to the tornado missile protection methodology for the EDG exhaust system, which resulted in a departure from a method of evaluation described in the USAR, without obtaining a license amendment. However, because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this Severity Level IV violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000263/2007006-01(DRS)). The licensee entered the finding into their corrective action program as AR01079705.

1R17 Permanent Plant Modifications (71111.17B)

.1 Review of Permanent Plant Modifications

a. Inspection Scope

From February 12, 2007, through March 1, 2007, the inspectors reviewed ten permanent plant modifications that had been installed in the plant during the last two years. This included two engineering changes, three equivalency evaluations, and five setpoint changes. The modifications were chosen based upon risk significance, safety significance, and complexity. As per inspection procedure 71111.17B, two modifications were chosen that affected the barrier integrity cornerstone. The inspectors reviewed the modifications to verify that the completed design changes were in accordance with the specified design requirements, and the licensing bases, and to confirm that the changes did not adversely affect any systems' safety function. Design and post-modification testing aspects were verified to ensure the functionality of the modification, its associated system, and any support systems. The inspectors also verified that the modifications performed did not place the plant in an increased risk configuration.

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.

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 February 12, 2007, through March 1, 2007, the inspectors reviewed 18 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 plant 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

The inspectors presented the inspection results to Mr. J. Grubb and others of the licensee's staff, on March 1, 2007. Licensee personnel acknowledged the inspection results presented. Licensee personnel were asked to identify any documents, materials, or information provided during the inspection that were considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
R. Baumer, Licensing
F. Domke, Electrical Design Supervisor
J. Grubb, Engineering Director
B. Guldemond, Nuclear Safety Assurance Manager
N. Haskell, Engineering Design Manager
T. Hurrle, Configuration Management Supervisor
D. Nordell, Configuration Management Engineer
J. Ohotto, Design Engineering Supervisor
D. Pennington, Design Engineer
B. Sawatzke, Plant Manager

Nuclear Regulatory Commission
D. Hills, Chief, Engineering Branch 1, Division of Reactor Safety
S. Thomas, Senior Resident Inspector
L. Haeg, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

05000263/2007006-01	NCV	Inadequate 10 CFR 50.59 Evaluation for Diesel Generator Exhaust Missile Protection (Section 1R21.3.b)

LIST OF DOCUMENTS REVIEWED

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

IR02 Evaluation of Changes, Tests, or Experiments 71111.02

10 CFR 50.59 Evaluations

SE-03-004; Diesel Exhaust Missile Protection Design Consideration; dated July 28, 2003

SE-06-003; SBO Operator Actions Associated with the HPCI System; dated September 19, 2006

10 CFR 50.59 Screenings

SCR-04-0283; SRV Air Actuator Model Change; dated November 23, 2005

SCR-04-0859; HPCI Turbine Steam Supply Low Pressure Isolation; dated September 11, 2006

SCR-05-0161; Set Point for RHR Minimum Flow Switches FS-10-2-121, A, B, C and D; dated August 23, 2006

SCR-05-0242; Instrument Setpoint Calculation 4.16KV Degraded Voltage; dated March 28, 2006

SCR-05-0266; ITS Setpoint Change - HPCI Steam Line Area Temperature – High; dated August 26, 2006

SCR-05-0689; Calc CA-05-146, Evaluation of Wall Thinning on FW2B-10-ED; dated October 11, 2005

SCR-05-0738; Calc CA-05-028, Evaluation of HPCI Condensate Drain Line D13-2"-HE in the HPCI Room; dated November 9, 2005

SCR-05-0739; Calc 05-147, Evaluation of HPCI Module E.2; dated November 9, 2005

SCR-05-0757; Chilled Water Expansion Tank V-CT-1 Replacement; dated November 15, 2005

SCR-05-0822; CA-05-155, Evaluation of Offgas Stack for SSE Seismic Loads; dated December 22, 2005

SCR-06-0062; Less than Full Thread Engagement on RWCU AO Valve Actuator Bonnet Nuts; dated February 15, 2006

SCR-06-0103; HPCI Steam Void Elimination; dated April 6, 2006

SCR-06-0106; Service Water Pump Replacement; October 30, 2006
SCR-06-0165; Replace AO-1575(6) and Check Valves with Normally Closed Valve SW-228(9); dated October 31, 2006

SCR-06-0166; Replace Rotork Actuators on Five MOVs with Limitorque Actuators; dated April 26, 2006

SCR-06-0310; Technical Requirements Manual - Appendix B - Secondary Containment Isolation Valves; dated September 12, 2006

SCR-06-0557; Suppression Chamber Inspection; dated December 4, 2006

SCR-07-0043; Fuel Storage and Handling Systems, Design Basis; dated January 22, 2007

<u>10 CFR 50.59 Applicability Determinations</u>

SCR-05-0645; Drawing Classification Level Change to '3'; dated September 19, 2005

SCR-05-0657; Combustible Loading Calculation; dated September 22, 2005

SCR-05-0663; Replace Fusible Link on V-DF-SBGT-2 with One of a Higher Temperature Rating; dated September 28, 2005

SCR-05-0791; Evaluation of Fire Detector Locations in the Reactor Building; dated December 5, 2005

SCR-05-0819; Setpoint Change Request for the Safety/Relief Valve Low-Low Set Logic to Incorporate the New Trip Settings; dated December 21, 2005

SCR-05-0830; Setpoint Change Request for the 4KV Bus-15 and Bus 16 Undervoltage Relays to Incorporate the New Trip Setting; dated January 3, 2006

SCR-06-0308; Update USAR for Improved Technical Specification Project; dated July, 29, 2006

<u>IR17</u> <u>Permanent Plant Modifications</u> 71111.17B

<u>Modifications</u>

EC8819; HPCI Steam Line Area Temperature – High; dated October 27, 2006

EC7583; Degraded Voltage Relays for Safety-Related 4KV Busses ; dated August 7, 2006

Equivalency Evaluations

EC910; Replacement Blower Wheel; Revision 1

EC933 (05A099); HPCI Auxiliary Lube Oil Pump; Revision 0

EC7828; Engine Driven Fuel Pump Suction Line; Revision 0

Setpoint Changes

EC8818; HPCI Turbine Steam Line Pressure - Low; dated October 27, 2006

EC8792; LPCI Pump Discharge Flow - Low; dated October 27, 2006

SCR 05-022; 4KV Bus-15 and Bus-16 Undervoltage Relays; dated December 1, 2005

SCR 05-023; Main Steam Line Steam Chase High Temp Group 1 Isolation; dated December 1, 2005

SCR 05-028; SRV Low Low Set Pressure Interlock; dated December 1, 2005

Other Documents Reviewed During Inspection

Corrective Action Program Documents Generated As a Result of Inspection

AR01076896; List to NRC Screened out All 50.59 Screening using the 3283 Form;

AR01077202; SCR-05-0830 Description Contains Incorrect Value; dated February 14, 2007

AR01077855; Action to Correct Drawing Error was Cancelled; dated February 19, 2007

AR01078665; Error in Calculation CA-05-146, Evaluation of Wall Thinning in FW2B-10"-ED; dated February 22, 2007

AR01079705; LAR Required for Use of TORMIS Code Methodology; dated February 28, 2007

AR01080049; SCR-05-0161 Activity Incorrectly Categorized; dated March 1, 2007

Corrective Action Program Documents Reviewed During the Inspection

AR00824446; NDE Thickness < 87.5 percent TNOM on FW2B-10"-ED, “B” Feedwater to Reactor Line; March 25, 2005

AR00891838; Evidence of Water Leakage on 11 and 12 EDG Exhaust Pipe Insulation; dated September 28, 2005

AR01000610; Replacement Part does not Match the Part Removed; dated October 10, 2005

AR01000746; Inconsistency Between Line Design Table and Plant; dated October 11, 2005

AR01001520; Operation past One Cycle Not Assured for Fw Pipe; dated October 20, 2005

AR01003632; RC-44-2 Replacement Noticed 3000 No. vs. 6000 No.; dated November 14, 2005

AR01004032; RWC Pipe Support Discrp and Indad Thread Engage on Act Nuts; dated November 17, 2005

AR01006064; CV-1728 Plug Replaced, No Section XI Repair/Replacement Plan; dated December 1, 2005

AR01008347; Some SW Mods May Inadvertently Create New Problems; dated December 21, 2005

AR01022687; SW 1-18"-JF Does Not Meet Class 1 Design Criteria ; dated April 6, 2006

AR01026395; Potential Exists for Failure to Manually Start ECCS Room Coolers; dated April 26, 2006

AR01040014; Inadequate Closeout Activities for Design Change 99Q160; dated July 17, 2006

AR01059716; Change to PM Frequency not Considered; dated November 3, 2006

AR01059908; Adverse Trend in Modification Implementation; dated November 6, 2006

AR00891237; No Column Gaskets Found on RHRSW Pump Columns; dated September 27, 2005

AR1040142; B.03.04-05 Issued Prior to Completion of Revision Process; dated July 18, 2006

AR0780295; Revise USAR Section 10.2.4.3 to Reflect the Results of CA-95-028; dated November 26, 2006

AR01045206; 50.59 Screening SCR-05-210 Missed USAR Impact; dated August 18, 2006

<u>Calculations</u>

CA-03-038; Instrument Setpoint Calculation, 4.16 KV Loss of Voltage; Revision 1

CA-03-039; Instrument Setpoint Calculation - SRV Low-Low Set, Reactor Coolant System Pressure; Revision 0

CA-04-110; Determination of HPCI Area High Temperature Setpoints; Revision 1

CA-05-108; Evaluation of Wall Thinning on FW2B-10-ED Piping; Revision 0

CA-05-146; Evaluation of Wall Thinning on FW2B-10"-ED Piping; Revision 0

Drawings

EC-811-01; Monticello Nuclear Generating Plant Installation of HPCI Void Resolution; Revision 1

NH-36250; Monticello Nuclear Generating Plant P&ID (Water Side) High Pressure Coolant Injection System; Revision AF

LIST OF ACRONYMS USED

ADAMS	Agency-Wide Document Access and Management System
AR	Action Request
CFR	Code of Federal Regulations
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
EC	Engineering Change
EPRI	Electric Power Research Institute
IMC	Inspection Manual Chapter
IR	Inspection Report
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PARS	Publicly Available Records
PRA	Probabilistic Risk Assessment
SCR	Screening (50.59)
SCR	Setpoint Change Request
SDP	Significance Determination Process
SE	Safety Evaluation (50.59)
TS	Technical Specifications
USAR	Updated Safety Analysis Report