June 16, 2000

Mr. Oliver D. Kingsley
President, Nuclear Generation Group
Commonwealth Edison Company
ATTN: Regulatory Services
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: BRAIDWOOD - NRC INSPECTION REPORT 50-456/2000007(DRS); 50-457/2000007(DRS)

Dear Mr. Kingsley:

On May 19, 2000, the NRC completed an inspection at your Braidwood Units 1 and 2 reactor facilities. The results of this inspection were discussed on May 19, 2000, with Mr. Tulon and other members of your staff. The enclosed report presents the results of that inspection.

The inspection was an examination of activities conducted under your license as they relate to 10 CFR 50.59 screenings and evaluations, the corrective action process relating to 10 CFR 50.59 discrepancies, and to compliance with the Commission's rules, regulations and the conditions of your license. Within these areas the inspection consisted of a selective examination of procedures and representative records and interviews with personnel. Specifically, this inspection focused on the performance of 10 CFR 50.59 under the Reactor Safety strategic performance area and the Initiating Events (IE), Mitigating Systems (MS) and Barrier Integrity (BI) cornerstones.

During this inspection no findings were identified.

In accordance with 10 CFR 2.790 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). ADAMS is accessible from* the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA by Steve Reynolds Acting For/

Ronald N. Gardner, Chief
Electrical Engineering Branch
Division of Reactor Safety

Docket Nos. 50-456; 50-457
License Nos. NPF-72; NPF-77

Enclosure: Inspection Report 50-456/2000007(DRS);
50-457/2000007(DRS)

cc w/encl: D. Helwig, Senior Vice President, Nuclear Services
C. Crane, Senior Vice President, Nuclear Operations
H. Stanley, Vice President, Nuclear Operations
R. Krich, Vice President, Regulatory Services
DCD - Licensing
T. Tulon, Site Vice President
K. Schwartz, Station Manager
T. Simpkin, Regulatory Assurance Supervisor
M. Aguilar, Assistant Attorney General
State Liaison Officer
Chairman, Illinois Commerce Commission

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-456, 50-457
License Nos:	NPF-72, NPF-77
Report No:	50-456/200007(DRS); 50-457/2000007(DRS)
Licensee:	Commonwealth Edison Company
Facility:	Braidwood Nuclear Plant, Units 1 and 2
Location:	35100 S. Route 53 Suite 84 Braceville, IL 60407-9617
Dates:	May 15 - 19, 2000
Inspector:	R. Winter, Reactor Engineer
Approved by:	Ronald N. Gardner, Chief Electrical Engineering Branch Division of Reactor Safety

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	Radiation Safety	Safeguards
● Initiating Events	● Occupational	● Physical Protection
● Mitigating Systems	● Public	
● Barrier Integrity		
● Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF FINDINGS

Braidwood Nuclear Plant, Units 1 and 2
50-456/2000007(DRS); 50-457/2000007(DRS)

The report covers a one-week announced inspection by a regional reactor engineer. This inspection assessed 10 CFR 50.59 screenings and evaluations selected from risk significant systems and covering activities such as modifications, procedure revisions, tests or non-routine operating configurations, changes to the USFAR, and calculations. The inspection also assessed the corrective action process relating to 10 CFR 50.59 discrepancies.

Initiating Events, Mitigating Systems and Barrier Integrity

- There were no findings identified.

Report Details

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems and Barrier Integrity

1R02 Changes, Tests or Experiments (Attachment 71111.02)

.1 Review of 10 CFR 50.59 Evaluations and Screenings for Changes, Tests or Experiments

a. Inspection Scope

The inspectors reviewed eleven evaluations done pursuant to 10 CFR 50.59 The majority pertained to the mitigating systems cornerstone. All eleven evaluations were appropriately analyzed and sufficiently documented. The inspectors also reviewed seventeen changes to the updated final safety analysis report where the licensee had determined that a 10 CFR 50.59 evaluation was not necessary. In regard to the seventeen changes where no 10 CFR 50.59 evaluation was performed, the inspectors verified that the changes did not meet the threshold of a "change to the facility as described in the safety analysis report" or a "test or experiment not described in the safety analysis report." For the 10 CFR 50.59 evaluations, the inspectors confirmed that prior NRC approval was not required for any of the changes.

b. Issues and Findings

No findings were identified in this area.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed condition reports concerning 10CFR50.59 issues to verify that the licensee had an appropriate threshold for identifying issues. The inspectors also evaluated the effectiveness of the corrective actions to the identified issues.

b. Issues and Findings

No findings were identified in this area.

4OA5 Management Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. Tulon and other members of licensee management at the conclusion of the inspection on May 19, 2000. The licensee acknowledged the results of the inspection. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

J. Bailey, Regulatory Assurance
S. Butler, Regulatory Assurance
M. Cassidy, Regulatory Assurance - NRC Coordinator
C. Dunn, Operations Manager
D. Lawson, Nuclear Oversight Assessor Manager
F. Lentine, Design Engineering Manager
D. Pierce, Training Instructor
K. Schwartz, Station Manager
T. Simpkin, Regulatory Assurance Manager
T. Tulon, Site Vice President

NRC

R. Gardner, Chief, Electrical Engineering Branch
C. Phillips, Senior Resident Inspector

INSPECTION PROCEDURES USED

IP 71111-02	Changes, Tests, or Experiments
IP 71152	Identification and Resolution of Problems (reference)

ITEMS OPENED, CLOSED AND DISCUSSED

Opened

None

Closed

None

Discussed

None

LIST OF ACRONYMS USED

BI	Barrier Integrity
BwAP	Braidwood Administrative Procedure
CFR	Code of Federal Regulations
DRS	Division of Reactor Safety
FSAR	Final Safety Analysis Report
IE	Initiating Events
MS	Mitigating Systems
NRC	Nuclear Regulatory Commission
NSP	Nuclear Station Procedure
PERR	(NRC) Public Electronic Reading Room

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.

Problem Identification Forms

A1998-00027	10 CFR 50.59 Tracking System Workaround
A1998-01197	Q&SA Identified Failure to Perform 10 CFR 50.59 Evaluations CAR 20-98-019
A1998-01602	Q&SA Identified - Weak 10 CFR 50.59 Screening for BwAP 1205-2T2, Revision 31
A1998-01619	1/2 BwGP 10001T5 10 CFR 50.59 Screening For Current Revision
A1998-01866	Q&SA Audit Concern - Adequacy of BwAP's on Software Control CAR 20-98-037
A1998-02385	Four persons missed 10 CFR 50.59 Refresher training - SITE ENG
A1998-02427	One no show for 10 CFR 50.59 refresher training
A1998-02428	One person - SGRP group missed 10 CFR 50.59 refresher training
A1998-02465	One engineering no show for 10 CFR 50.59 refresher training
A1998-02520	Inadequate preparations for new 10 CFR 50.59 procedure
A1998-02540	One IMD person missed 10 CFR 50.59 refresher training
A1998-02541	One Site Engineering person missed 10 CFR 50.59 refresher training
A1998-02562	One Health Physics person did not show for 10 CFR 50.59 refresher
A1998-02725	Failure to complete 10 CFR 50.59 in a timely manner
A1998-02726	Failure to complete 10 CFR 50.59 in a timely manner
A1998-02728	Failure to complete 10 CFR 50.59 in a timely manner
A1998-02729	Failure to complete 10 CFR 50.59 in a timely manner
A1998-02730	Failure to complete 10 CFR 50.59 in a timely manner
A1998-02731	Failure to complete 10 CFR 50.59 in a timely manner
A1998-03654	Failure to submit 10 CFR 50.59 Summaries for 10 CFR 50.59 Validations
A1998-04376	NO identified the lack of a 10 CFR 50.59 for sea vans used to store RAM
A1999-00217	NO identified failure to follow QA manual for transmittal of 10 CFR 50.59s to Nuclear Oversight

A1999-00436	Long-Term OOS 10 CFR 50.59 Screenings/Evaluations not completed
A1999-00561	Safety Evaluation Validation does not meet expectations/Standards
A1999-00779	Lost information on 10 CFR 50.59 tracking in ECF following restoration of System
A1999-00789	QRT Review of BRW-SE-1999-27 Requires Revision
A1999-00805	QRT Review of BRW-SE-1999-43
A1999-00812	Deficiencies Identified in Safety Evaluation Validation BRW-SESV-1999-83 by QRT
A1999-00856	QRT Review of BRW-SE-1999-12
A1999-00858	QRT Review of BRW-SESV-1999-125
A1999-01025	QRT Review of BRW-PTES-1999-221
A1999-01133	10 CFR 50.59 Screening for TMOD 99-0-006 Addresses Wrong System
A1999-01238	QRT Review of BRW-SE-1999-356
A1999-01294	QRT Review of BRW-SE-1999-347
A1999-01746	QRT Review of BRW-PTES-1999-412
A1999-01848	10 CFR 50.59 Qualification Documentation not Complete
A1999-01958	NO Identified that 10 CFR 50.59 Evaluation was not performed for Unit 1 Calorimetric Program
A1999-01959	NO Identified that 10 CFR 50.59 Evaluation was not performed for Unit 2 Calorimetric Program
A1999-02005	QRT required revision to 10 CFR 50.59, BRW-SE-1999-663
A1999-02047	NO identified incorrect use of 10 CFR 50.59 validation
A1999-02171	QRT review of 10 CFR 50.59 Screening BRW-PTES-1999-707
A1999-02172	QRT Review of Safety Evaluation BRW-SE-1999-711
A1999-02188	QRT Review of SE BRW-SE-1999-725, MSIV Room Roof Hatches
A1999-02200	QRT Review of SE BRW-SE-1999-709, Adding Vent Valves & Lines to ECCS Piping
A1999-02257	Nuclear Oversight Identified that Safety Evaluation BRW-SE-1999-802 is Incomplete
A1999-02947	QRT Review of Safety Evaluation BRW-SE-1999-846
A1999-02968	QRT Review-10 CFR 50.59 BRW-PTES-1999-973 requires revision
A1999-03171	10 CFR 50.59 screening found to be of poor quality
A1999-03346	NO Failure to communicate Potential issues to Engineering
A1999-03864	10 CFR 50.59 screening is not conducted for procedure editorial changes
A1999-03365	QRT review of BRW-PTES-1999-857
A1999-03821	Inappropriate changes to PEPP-E Forms for 10 CFR 50.59 Safety Evaluations
A1999-03975	Previously deleted 10 CFR 50.59 form was used for a screening
A2000-00687	Long-Term OOS 10 CFR 50.59 evaluations not completed on time
A2000-00811	Step box not marked on Safety evaluation
A2000-01869	Review of 10 CFR 50.59 Safety Evaluation Screening
A2000-02224	At Dresden incorrectly performed 10 CFR 50.59, identified causes & corrective actions

Procedures

BwAP 1205	Braidwood Station 10 CFR 50.59 Processing, Revision 7E1
BwAP 1245-1	Instructions for UFSAR/FPR Revision and Treatment of Possible Discrepancies, Revision 7
NSP-AP-4004	Corrective Action Program Procedure, Revision 4
NSP-CC-3001	Operability Determination Process, Revision 0
NSP-CC-3011	UFSAR and Fire Protection Report Update Procedure, Revision 1, November 12, 1998
RS-AA-104	10 CFR 50.59 Safety Evaluation Process, Revision 0, August 9, 1999

10 CFR 50.59 Screenings and Evaluations

SCREENED OUT

BRW-PTES-1999-511	Screening of vendor procedure, Westinghouse MRS 2.4.2 GEN 45, Revision 0, ABB In - Situ Pressure Testing for SG tubes"
BRW-PTES-1999-705	Change scale inhibitor at Lake Screen House for CW, SX, WS sys NALCO 9249 Polyacralyte to NALCO Sure-Cool 1393-HEDP
BRW-PTES-1999-778	Changed the Pressurizer Level Setpoint Control Program from 29-60% to 26-60% & S/G level on scale setpoint from 11 to 14%
BRW-PTES-1999-836	Procedure ER-AA-370, "Ferritic Reactor Coolant Pressure Boundary (FERCPB) Integrity Program"
BRW-PTES-1999-1193	Change incorporates time for slave relay actuation into overall ESF response time compilation, previously excluded
BRW-PTES-1999-1318	New OAD procedures for performance of switchyard 345KV Bus and Bus-Tie breaker protective relay routines
BRW-PTES-1999-1394	Revision of acceptance criteria for BwVS 900-29 testing of the U-1 component cooling heat exchanger adjusted for tubes plugged
BRW-PTES-2000-89	Revision to 1BwVSR 5.5.8.SX.2, "ASME Surveillance Requirements for 1B Essential Service Water Pump," Revision 01
BRW-PTES-2000-178	BwVSR 5.5.8.SI.3, Revision 2, SI Sys Check Valve Stroke Test to allow stopwatch and/or stripchart recorder for response testing"
BRW-PTES-2000-324	Revise ER-BR-540-001, Revision 1, "U-1 Diesel driven auxiliary FW Pump performance Test and control valve throttle setting verification"
BRW-PTES-2000-339	Revise BwOP SI-5, Revision 16, "Expand procedure to include an option to fill accumulators from the "B" SI pump"
BRW-PTES-1999-363	Update 2BwOA RCP-1 to increase the allowable seal leakoff flow for an immediate Reactor trip and RCP shutdown from 6 to 8 gpm
BRW-PTES-1999-369	Change procedures - Change Unit 2 to mirror Unit 1 changes for AT&T battery changeout
BRW-PTES-1999-391	Change minimum pilot cell temperature from 40 F to 65 F and min temperature of water to be added to battery cells in Limitation
BRW-PTES-1999-415	Procedure BwVP 500-8, "Rod Drive troubleshooting Precaution Guide used during troubleshooting of Westinghouse RD system"
BRW-SESV-2000-235	Revise 1/2BwOSR 3.5.5.1 to reflect requirements of Tech Spec 3.5.5 to allow usage of average PZR pressure

BRW-FCS-1999-1269	SSCRs 99-035 & 99-036 U-1 and U-2 FW Pump NPSW Low Alarm Deadband Change

FULL 10 CFR 50.59 EVALUATIONS

BRW-SESV-1999-318	Revised the original discussion of the setpoint at which AF pump suction switchover from CST to SX occurs
BRW-SESV-2000-341	Special Test SPP-00-006 satisfies the IST full stroke open exercise test requirement for the 1A AF Pump discharge check valve 1AF029A by passing the maximum required accident condition flow
BRW-SE-1999-429	SPPs for connecting and disconnecting a Temporary Battery to each U-2 125 V DC ESF Bus. The temporary Battery will be connected to the 125 V ESF DC bus for a maximum of 10 days per train to support on-line installation of new C&D batteries
BRW-SE-1999-591	Provide an alternate source of 480Vac power to the constant voltage transformers (CVTs) for Instrument Bus 212 (2IP02E) and 214 (2IP04E) from Turbine Building MCC 234Y3
BRW-SESV-1999-799	Permanent installation of ultrasonic fw flow transducers installed on the U-1 and U-2 fw lines under temporary Modifications (TMOD) 99-1-010 and 98-2-015
BRW-SE-1999-1066	Trip reduction program design change. Revise logic control for the Turbine Intercept Valves (IVs) to eliminate the wiring for the Close Intercept Valve (CIV) function associated with the Turbine Digital Electro-Hydraulic Computer (DEHC). The DRP revises the UFSAR to delete descriptions of the LDA and CIV.
BRW-SE-1999-1194	DCP 9800282 and 283 will upgrade the RC Pump motor protection logic by replacing the model CO-11 with CO-8 overcurrent relays
BRW-SE-2000-0046	Freeze Seals for valves 2SX052B & 2SX057B for 10' lines service water piping
BRW-SE-2000-132	Revise Technical Specification Bases B 3.9.2. The proposed Bases Change provides clarification of potential (RWST) dilution source flowpaths.
BRW-SESV-2000-195	1/2BwOAPRI-10,Revision 58. Revise procedure to allow emergency boration via an SI pump or CV pump when the RCS is depressurized.
BRW-SE-2000-217	Revises the cycling setpoints lower, the compressor start/stop cycle starts at 225 psig and stops at 235 psig.