October 13, 2008

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

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2
EVALUATIONS OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000456/2008008(DRS); 05000457/2008008(DRS)

Dear Mr. Pardee:

On September 12, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed the evaluations of changes, tests, or experiments and permanent plant modifications inspection at your Braidwood Station, Units 1 and 2. The enclosed report documents the inspection results, which were discussed on September 12, 2008, with Mr. Larry Coyle and other members of your staff.

The inspection 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 this inspection, no findings of significance were identified.

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 (PARS) component of

NRC's document system (ADAMS). ADAMS is accessible from the NRC Website 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 Nos. 50-456; 50-457
License Nos. NPF-72; NPF-77

Enclosure: Inspection Report 05000456/2008008(DRS); 5000457/2008008(DRS)
w/Attachment: Supplemental Information

cc w/encl: Site Vice President - Braidwood Station
Plant Manager - Braidwood Station
Regulatory Assurance Manager - Braidwood Station
Chief Operating Officer and Senior Vice President
Senior Vice President - Midwest Operations
Senior Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
Director - Licensing and Regulatory Affairs
Manager Licensing - Braidwood, Byron and LaSalle
Associate General Counsel
Document Control Desk - Licensing
Assistant Attorney General
J. Klinger, State Liaison Officer,
Illinois Emergency Management Agency
Chairman, Illinois Commerce Commission

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Associate General Counsel
Document Control Desk - Licensing
Assistant Attorney General
J. Klinger, State Liaison Officer,
Illinois Emergency Management Agency
Chairman, Illinois Commerce Commission

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Letter to Mr. Charles G. Pardee from Mr. David E. Hills dated October 13, 2008.

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2
EVALUATIONS OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000456/2008008(DRS); 05000457/2008008(DRS)

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REGION III

Docket Nos:	50-456; 50-457
License Nos:	NPF-72; NPF-77
Report Nos:	05000456/2008008(DRS) and 05000457/2008008(DRS)
Licensee:	Exelon Generation Company, LLC
Facility:	Braidwood Station, Units 1 and 2
Location:	Braceville, IL
Dates:	August 25, 2008, through September 12, 2008
Inspectors:	George M. Hausman, Senior Reactor Inspector (Lead) John V. Bozga, Reactor Inspector Benny Jose, Senior Reactor Inspector
Observer:	Christian B. Scott, Reactor Engineer
Approved by:	David E. Hills, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000456/2008008(DRS); 05000457/2008008(DRS); 08/25/2008 - 09/12/2008; Braidwood Station, Units 1 & 2; Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications.

The inspection covered a two-week announced baseline inspection on evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by three regional based engineering inspectors. Based on the results of this inspection, no findings of significance were identified. 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: Initiating Events

No findings of significance were identified.

Cornerstone: Mitigating Systems

No findings of significance were identified.

Cornerstone: Barrier Integrity

No findings of significance were identified.

B. Licensee-Identified Violations

No findings of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

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

.1 Evaluations of Changes, Tests, or Experiments

a. Inspection Scope

From August 25, 2008, through September 12, 2008, the inspectors reviewed 10 evaluations 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 also reviewed 22 screenings where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. In regard to the changes reviewed where no 10 CFR 50.59 evaluation was performed, the inspectors verified that the changes did not meet the threshold to require a 10 CFR 50.59 evaluation. The evaluations and screenings were chosen based on risk significance, safety significance, and complexity. Documents reviewed are listed in the attachment to this report.

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 constitutes 10 samples of evaluations and 22 samples of changes as defined in Inspection Procedure 71111.17-05.

b. Findings

During this inspection, the NRC senior resident inspector requested the team's assistance with their review of the 10 CFR 50.59 evaluation EC361637 (FDRP 23-003) "Abandon the Upper Cable Spreading Room Carbon Dioxide (CO_2) System," Revision 0. The results of that review will be documented in the Braidwood Integrated Inspection Report 2008004.

.2 Permanent Plant Modifications

a. Inspection Scope

From August 25, 2008, through September 12, 2008, the inspectors reviewed 13 permanent plant modifications that had been installed in the plant during the last three years. The modifications were chosen based upon risk significance, safety significance, and complexity. As per Inspection Procedure 71111.17, one modification was chosen that affected the design bases and functioning of interfacing systems as well as introducing the potential for common cause failures. 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.

This inspection constitutes 13 samples as defined in Inspection Procedure 71111.17-05.

b. Findings

(1) Temporary/Permanent Conversion of Lead Shielding on Piping Systems

Introduction: The inspectors identified an unresolved item (URI) concerning seismic Category I pipe supports associated with the safety injection (SI) and residual heat removal (RH) piping systems. Design documents for the 2SI06 and 2RH01 piping subsystems' pipe supports were not sufficiently detailed to demonstrate compliance with the American Institute of Steel Construction (AISC) Manual of Steel Construction Code and the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

Description: The SI and RH piping systems are part of the emergency core cooling system (ECCS). The Braidwood Updated Final Safety Analysis Report (UFSAR), Section 6.3.1, stated the primary function of the ECCS is to remove the stored and fission product decay heat from the reactor during accident conditions. The ECCS also provides shutdown capability for design basis accidents by means of boron injection. The ECCS was classified as a safety class II system designed to meet seismic category I requirements.

The inspectors reviewed Calculation BRW-97-0827-M, "Piping Evaluation for Lead Shielding Installation on Subsystem 2SI06 Piping per Temporary Lead Shielding Request (TSR) No. 95-153, 96-018, 96-045, 96-053, and 97-120," Revision 0 and Minor Revision 0A. The purpose of Revision 0 was to evaluate the affect of the temporary lead shielding installed on the 2SI06 piping subsystem (i.e., lead shielding installed on sections of pipelines 2RH01BA-12", 2RH01BB-12", 2RH01CA-16", 2RH01CB-16", 2SI06BA-24", and 2SI06BB-24") by the TSRs. The purpose of Minor Revision 0A was to evaluate the affect of converting the temporary lead shielding to permanent lead shielding. In addition, Minor Revision 0A identified that based on recent industry concerns the lead shielding weighed up to 10 percent more than was previously analyzed.

Pipe stresses were determined from loads and load combinations due to internal pressure, pipe system dead weight, pipe thermal expansion and seismic excitation. The 2SI06 and 2RH01 piping subsystems were designed to the ASME Boiler and

Pressure Vessel Code, Section III, Subsection NC and ND, 1977 Edition up to and including the 1979 Addenda. The associated pipe supports were designed to the AISC Manual of Steel Construction Code and the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, 1977 Edition through Summer 1979 Addenda. The seismic response spectra analysis of the piping subsystem was analyzed using the ASME Code Case N-411, "Alternative Damping Values for Seismic Analysis of Classes 1, 2, and 3 Piping Sections, Section III, Division 1." As specified in the licensee's UFSAR, Section 3.7, Table 3.7-1, "Damping Values," the ASME Code Case N-411 may be used for alternative damping values when NRC conditions as defined in Regulatory Guide 1.84, "Design and Fabrication Code Case Acceptability ASME Section III Division I," Revision 24 are met.

In Calculation BRW-97-0827-M, Revision 0, the licensee's qualification of the pipe supports were based on their review of Calculation 13.2.29, "Structural Calculation for Mechanical Component Support [*Pipe Support Number*]," Revision 2 and the use of engineering judgment. The licensee concluded that sufficient margin existed in the pipe support design such that the supports would be able to withstand the increased loads from the installed lead shielding.

The inspectors reviewed the Structural Calculation for Mechanical Component Support [*Pipe Support Number*] contained in Calculation 13.2.29, Revision 2, for the following:

Pipe Support Number	*Pipe Support Number*	*Pipe Support Number*
2SI06309X	2SI06328X	2SI06342X
2SI06310X	2SI06335X	2SI06345X
2SI06311G	2SI06336X	2SI06351X
2SI06316X	2SI06337X	2SI06358X
2SI06318X	2SI06340S	2SI06360X

The inspectors determined that the engineering judgment used in Calculation BRW-97-0827-M, Revision 0 was not valid and the aforementioned pipe supports could exceed their design basis and operability acceptance limits. Also, a condition specified in ASME Code Case N-411 stated "This Code Case is not appropriate for analyzing the dynamic response of piping systems using supports designed to dissipate energy by yielding." The licensee initiated issue report (IR) 00816677, "NRC MOD/50.59 Inspection - 2SI06 Piping Subsystem Support," dated September 11, 2008, to address this issue.

In response to IR 00816677, the licensee's prompt operability determination concluded through analysis that the fillet weld connection between the attachment plate and the embedment plate for pipe support 2SI06316X exceeded design basis and operability limits. Subsequently, the licensee performed a walkdown to field verify the actual fillet weld size of this connection. The actual fillet weld size was determined to be 7/16" thick, which was greater than the 1/4" thick fillet weld size used in the analysis. The licensee determined that the pipe support 2SI06316X fillet weld connection exceeded the design basis limits but was within operability acceptance limits.

Further analysis by the licensee showed pipe supports 2SI06328X, 2SI06340S, 2SI06345X and 2SI06358X exceeded their design basis limits. The concrete expansion anchor bolt evaluation for each pipe support resulted in a factor of safety of less than four but greater than two. A factor of safety of greater than two satisfies the operability requirements specified in procedure OP-AA-108-115, "Operability Determinations (CM-1)," Revision 6.

Pipe support 2SI06318X was determined to have no margin for design basis acceptance limits and pipe support 2SI06337X had minimal design basis margin remaining. The licensee determined that the pipe supports 2SI066309X, 2SI06310X, 2SI06311G, 2SI06335X, 2SI06336X, 2SI06342X, 2SI06351X and 2SI06360X met design basis and operability acceptance limits.

This issue is considered unresolved pending the licensee's response to address the inspectors' request for additional information (RAI) regarding the following inspectors' concerns:

- The licensee's prompt operability determination was made without incorporating the additional 10 percent weight and installed locations of the temporary lead shielding identified by Calculation BRW-97-0827-M, Minor Revision 0A into the licensee's pipe stress computer analysis program. These changes have the potential to affect all pipe supports and anchors on the 2SI06 and 2RH01 piping subsystems.
- There are approximately 58 pipe supports and five pipe anchors associated with the 2SI06 piping subsystem, which were not verified to determine their design basis acceptability.
- Pipe supports and anchors for the 2RH01 piping subsystem were never verified to determine their design basis acceptability.
- A condition specified in the ASME Code Case N-411 states "When used for reconciliation work or for support optimization of existing designs, the effects of increased motion on existing clearances and on line mounted equipment should be checked." This condition along with the other four ASME Code N-411 specified conditions were not addressed in the pipe stress analysis for either the 2SI06 or 2RH01 piping subsystems.

At the end of this inspection, the licensee stated that the SI and RH piping systems' pipe stress computer analysis program would be revised to reflect the RAI concerns and the results provided to the NRC for review within three months of September 12, 2008.
(URI 05000457/2008008-01(DRS))

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Condition Reports

a. Inspection Scope

From August 25, 2008, through September 12, 2008, 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 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 team are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On September 12, 2008, the inspectors presented the inspection results to Mr. Larry Coyle and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

No interim exits meetings were conducted.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

L. Coyle, Plant Manager
C. Furlow, Design Engineering
G. Golwitzer, Regulatory Assurance
J. Gosnell, Design Engineering
D. Gustofson, Design Engineering
D. Ibrahim, Design Engineering
J. Knight, Nuclear Oversight
T. McCool, Operations
J. Morales, Design Engineering
R. Gadbois, Maintenance Director
J. Odeen, Project Management Director
J. Petty, Regulatory Assurance
D. Riedinger, Design Engineering Manager
B. Schipiour, Work Management Director
M. Smith, Engineering Director

Nuclear Regulatory Commission

B. Dickson, Senior Resident Inspector
A. Garmoe, Resident Inspector
J. Heath, Resident Inspector
M. Perry, Illinois Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000457/2008008-01(DRS)	URI	RAI To Determine Adequacy of Pipe Supports Designed for Design Basis Loading Conditions with Lead Shielding Installed on SI and RH Subsystems (Section 1R17.2b.(1))

Closed and 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.

CALCULATIONS

Number	Description or Title	Date or Revision
051336	Addendum Piping Stress Rpt ESW Sys 2SX13	003D
051790	Piping Stress Rpt for Rx Coolant PZR Sys 2RY09	00E
062890	Doc of Qual Calcs for Class 1 Supports in Piping Sys 2RY09-PZR Safety & Relief Vlvs	000B
067084	Doc of Qual Calcs for Class 1 Supports in Piping Subsystem 2SI25 Boron Injection Sys	01A
6.5.7-BYR02-064	Check Structural Adequacy of New Removable Platform (walkway) in U2 Refueling Pool	1
13.1.31-BRW-07-0099-S	Structural Eval of Existing Pipe Support M-1SI21047G for Revised Loads	0
13.1.34-BRW-06-0182-S	Eval of Support 1SX66007X for Revised Loads	0
13.1.34-BRW-06-0183-S	Eval of Support 1SX66006R for Revised Loads	0
13.1.34-BRW-06-0184-S	Eval of Support 1SX66009R for Revised Loads	0
13.2.29	Structural Calculations for the Following Mechanical Component Supports: 2SI06309X 2SI06328X 2SI06342X 2SI06310X 2SI06335X 2SI06345X 2SI06311G 2SI06336X 2SI06351X 2SI06316X 2SI06337X 2SI06358X 2SI06318X 2SI06340S 2SI06360X	2
13.2.31-BRW-07-0119-S	Qual of Existing Pipe Support M-2SI25011R	0
13.2.31-BRW-07-0120-S	Qual of Existing Pipe Support M-2SI25012X	0
13.2.31-BRW-07-0121-S	Qual of Existing Pipe Support M-2SI25013X	0
13.4.11.2-BRW-07-0127-S	Structural Eval of Existing Support M-1PSEH024S001T	0
14.1.14-BRW-2007-0031-S	Qual of Pipe Supports M-1ES35043R, M-1ES35044R, M-1ES35045R & M-1ES35046R	0
14.1.14-BRW-2007-0032-S	Qual of Pipe Supports M-1ES37047D & M-1ES38048D	0
14.2.1-BRW-07-0066-S	Qual of Pipe Support M-2ABF22021G	0
14.2.35-BRW-07-0011-S	Qual of Pipe Support 2WXF153020T	1
19-T-6	DG Loading During LOOP/LOCA	6A
24.2.1	Mech Component Supports-Design Control Summary for Plant Mod & Station Support Work	0
ATD-0026	Combustible Fire Loads	9-08
BRW-07-0003-M	Seismic Qual of 1.5” & 2” SI Throttle Vlvs (U1 & 2	0

CALCULATIONS

Number	Description or Title	Date or Revision
	Tag Nos. SI8810A-D, SI8816A-D & SI8822A-D)	
BRW-07-0102-M	Eval/Characterization of Through Wall Leakage from Line 2SX27DA-10 per Code Case N-513-1	0
BRW-97-0827-M	Piping Eval for Lead Shielding Installation on Subsystem 2SI06 Piping per Temp Lead Shielding Request TSR95-153, 96-018/045/053 & 97-120	0
BRW-97-0827-M	Piping Eval Lead Shielding on Subsystem 2SI06	0A
NED-I-EIC-0004	PZR Press Protection Channel Error Analysis	4
SITH-1	RWST Level Setpoints	6 & 7
SG-BRW-96-414-M	Piping Stress Rpt for Subsystem 1FW02	02A
SG-BRW-96-415-M	Piping Stress Rpt for Subsystem 1FW03	03A
SG-BRW-96-416-M	Piping Stress Rpt for Subsystem 1FW04	02A
SG-BRW-96-429-M	Piping Stress Rpt for Subsystem 1SD26	02A
SG-BRW-96-430-M	Piping Stress Rpt for Subsystem 1FW05	02A
SG-BRW-96-431-M	Piping Stress Rpt for Subsystem 1SD27	01A
SG-BRW-96-432-M	Piping Stress Rpt for Subsystem 1SD28	03A
SG-BRW-96-433-M	Piping Stress Rpt for Subsystem 1SD29	02A

CORRECTIVE ACTION PROGRAM DOCUMENTS ISSUED DURING INSPECTION

Number	Description or Title	Date or Revision
00811051	NRC Inspection Calc Request – Revisions Unavailable	August 26, 2008
00815101	NRC Mod/50.59 Insp – Rev 2 to BWOP CV-16 Did Not Specify Limitation Procedure Use to One Operating Cycle	September 8, 2008
00815166	Minor Discrepancies in Fire Combustible Loading Calc	September 8, 2008
00815707	NRC Mod/50.59 Insp – Issue with BRW-E-2007-122	September 9, 2008
00816038	NRC Mod/50.59 Insp – Calc Assumption	September 10, 2008
00816539	NRC Mod/50.59 Insp - Calc Discrepancy	September 11, 2008
00816619	NRC Mod/50.59 Insp - Concern Actions for EDG Freq Issue	September 11, 2008
00816677	NRC Mod/50.59 Insp - 2SI06 Piping Subsystem Support	September 11, 2008
00817175	NRC Observations Following Mod/50.59 Inspection	September 12, 2008

CORRECTIVE ACTION PROGRAM DOCUMENTS ISSUED PRIOR TO INSPECTION

Number	Description or Title	Date or Revision
00629351	CDBI FASA - DG Freq Variations Not Addressed in Calcs	May 14, 2007
00641501	Missing M&TE Selection Sheets for Calibration	June 18, 2007

DRAWINGS

Number	Description or Title	Date or Revision
20E-1-4012A	Key Diagram 120Vac Instrument Bus 111 (1IP01J) ESF Division II – Chanel I	P
20E-1-4030IP01	Schematic Diagram 7.5kVA Fixed Freq Inverter for Instrument Bus 111 (1IP05E)	N

50.59 EVALUATIONS

Number	Description or Title	Date or Revision
BRW-E-2007-122	Implement Results of RWST Vortex Testing	July 25, 2007
BRW-E-2007-147	Modifications to SI Throttle Vlvs 1SI8810A-D, 1SI8816A-D, 1SI8822A-S	0
BRW-E-2007-161	Change In-Core Decay Time for A1R13 from 100Hrs to 78Hrs	0
BRW-E-2007-168	Alternate Load Paths for Heavy Loads for A1R13	0
BRW-E-2007-221	Revise BwOPCV-16 to Temporarily Allow PZR Aux Spray Operation	0
BRW-E-2008-2	Modifications to the SI Throttle Vlvs 2SI8810A-D, 2SI8816A-D, 2SI8822A-D	0
BRW-E-2008-77	Alternate Load Paths for Heavy Loads for A2R13	0
BRW-E-2008-105	Operability Eval 07-008, Related to Reduction in Operator Action Time Required to Prevent Overfilling of a Ruptured SG During an SGTR Event	0
BRW-E-2008-135	Power Ascension Testing of New Digital Control Sys	0
EC361637 (FDRP 23-003)	Abandon the UCSR Carbon Dioxide (CO_2) Sys	0

MODIFICATIONS

Number	Description or Title	Date or Revision
EC0000356371	Reactor Head Hoist Upgrade	0
EC0000358829	ECCS Sump Screen Project to Resolve GSI 191 Replace Fiber Insulation w/MRI - SG & Piping Below EL 429ft	2
EC0000358934	U2 MCR Safety Related Recorder Replacements	0
EC0000359960	Rewire Seq of Events Recorder Cabinet to Correct Power Supply Failure Alarm	0
EC0000360141	Replace SI Throttle Vlv Trim, Bonnet, Stem, & Manual Operators & Elimination of Downstream Orifice Plates to Support GSI 191 (U1)	3
EC0000360143	Replace SI Throttle Vlv Trim, Bonnet, Stem, & Manual Operators & Elimination of Downstream Orifice Plates to Support GSI 191 (U2)	2
EC0000362458	Reactor Head Hoist Upgrade	0
EC0000363113	Replace 480V Feed Breakers for MCC Dist Panels for	0

MODIFICATIONS

Number	Description or Title	Date or Revision
	Coordination	
EC0000363956	Setpoint Change for 1CV8119 Relief Vlv	1
EC0000365465	Replace U1 Barton Transmitter EQ Seals in Containment	2
EC0000366583	Implement Results of RWST Vortex Testing	August 31, 2007
EC0000366605	Implement Results of RWST Vortex Testing U1 & U2 RWST Empty Level Change from 7 - 9 percent	July 27, 2007
EC0000366628	Revise U1 PZR Ch Dev Alarm from 2 - 3 percent of Full Power	0

OPERABILITY EVALUATIONS

Number	Description or Title	Date or Revision
07-005	Through-Wall Leakage of Line 2SX27DA-10	0

PROCEDURES

Number	Description or Title	Date or Revision
1BCA-1.1	Loss of Emergency Coolant Recirculation U1	105
1BwCA-1.1	Loss of Emergency Coolant Recirculation U1	200
1BCA-1.3	Sump Blockage Control Room Guideline U1	2
1BEP ES-1.3	Transfer to Cold Leg Recirculation U1	105
1BwCA-1.3	Sump Blockage Control Room Guideline U1	200
1BwEP ES-1.3	Transfer to Cold Leg Recirculation U1	200
BwOP CV-16	PZR Aux Spray Operation	2
CC-AA-102	Design Input & Config Change Impact Screening	15
CC-AA-103	Config Change Control for Perm Phy Plant Changes	17
CC-AA-309	Control of Design Analysis	7
CC-AA-309-1001	Guidelines for Prep & Processing Design Analyses	4
CC-AA-309-1011	General Station Piping Analysis	2
OP-AA-108-115	Operability Determinations (CM-1)	6

REFERENCES

Number	Description or Title	Date or Revision
53904	Equivalency Eval: Replacement of Obsolete Westinghouse SSPS Input Relays with Potter & Brumfield Relays	0
56327	Equivalency Eval: Replacement of Certain Potter & Brumfield Relays with Magnecraft Relays	0
61168	Equivalency Eval: Replacement of CR Vent Fan Mtrs with Mtrs that Have Vacuum Press Impregnated Insulation Sys	0
61753	Equivalency Eval	May 2, 2008
ASME Code Case	Eval Criteria for Temp Acceptance of Flaws in Moderate	March 28, 2001

REFERENCES

Number	Description or Title	Date or Revision
N-513-1	Energy Class 2 or 3 Piping Sect XI, Div 1	
BwSD-FP-23	U1 Sys Demonstration Halon 1301 Fire Protection	0
BwSD-FP-63	U2 Sys Demonstration Halon 1301 Fire Protection	0
EC369677	Tech Eval Documenting Effect of EDG Freq Variations on ECCS Pump Flows.	0
EC366629	Tech Eval to Evaluate CV, RH, & SI Pmp Performance Due to EDG Freq Shift Post-LOCA (59.2-60.8 Hz)	0
RG 1.84	Design & Fab Code Case Acceptability ASME Sect III Div I	24

50.59 SCREENINGS

Number	Description or Title	Date or Revision
BRW-S-2007-005	Design Change EC363874, TS Basis Change 07-001, TRM Change 07-001, FDRP 23-007 & UFSAR Change DRP 12-008	0
BRW-S-2007-034	EC363956, Setpoint Change for 1CV8119 Relief Vlv	0
BRW-S-2006-056	Permanently Remove Existing Ladder & Associated Diagonal Braces at U1 Containment Equip Hatch	0
BRW-S-2007-061	TRM Change Request 07-004	0
BRW-S-2007-066	Revise NED – MSD-032; Byron & Braidwood CR Heat Load Calc	0
BRW-S-2007-084	Movement of Heavy Loads in the Fuel Bldg & Spent Fuel Pool in Support of the Rack G Repair Project	0
BRW-S-2007-091	Turb Governor & Throttle Vlv Testing Freq Change	0
BRW-S-2007-093	Temp Remove Strut 2SX13002X from 2SX27DA-10 Install Leak Mitigation Clamp w/ ½” Vlv Tap Over Pinhole Leak	0
BRW-S-2007-099	Rx Trip or SI (EP-0) Response to Nuclear Power Generation/ATWS(FR-S.1)	0
BRW-S-2007-120	Runway Beams Extension for 2HC22G in U2 Containment	0
BRW-S-2007-124	Increase PR Ch Dev Setpoint to Prevent Nuisance Alarms	0
BRW-S-2007-137	New Lifting Lug at the U1 & U2 Containment Equip Hatch Penetration Sleeves EC366588 & EC366589	0
BRW-S-2007-149	U2 Containment Refueling Pool Platform	0
BRW-S-2007-153	Revise Power Dissension Procedures Due to Replacement of DEH Control Sys	0
BRW-S-2007-160	Revise Aux Bldg HVAC Sys Shutdown Procedure	0
BRW-S-2007-176	Fuel Bldg Crane Heavy Load Lifts >2000lbs & <30,000lbs Using the Aux Hook/BwFP FH-20T4	0
BRW-S-2007-177	Fuel Bldg Crane Heavy Load Lifts Using the 125 Ton Main Hook/BwFP FH-20T5	0
BRW-S-2007-207	Revise DG Alignment to Standby Condition Procedure	0
BRW-S-2008-075	Operation with Higher RH Letdown Flow Rates	0
BRW-S-2008-090	Eliminate Snubber 2RY09060S on PZR PORV Line to	0

50.59 SCREENINGS

Number	Description or Title	Date or Revision
	Eliminate Interferences During WOL Installation	
BRW-S-2008-096	UFSAR Change Describe Additional Analysis for CS Pmps	0
BRW-S-2008-119	Adverse Cooling Lake Conditions	0

LIST OF ACRONYMS USED

ADAMS	Agency Wide Document Access and Management System
AISC	American Institute of Steel Construction
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CNO	Chief Nuclear Officer
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
IMC	Inspection Manual Chapter
IR	Inspection Report and Issue Report
MOD	Modification
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OA	Other Activities
PARS	Publicly Available Records System
RAI	Request for Additional Information
RH	Residual Heat Removal
ROP	Reactor Oversight Process
SDP	Significance Determination Process
SI	Safety Injection
TS	Technical Specifications
TSR	Temporary Lead Shielding Request
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item