UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 E. LAMAR BLVD.
ARLINGTON, TX 76011-4511

April 10, 2014

Jeremy Browning, Site Vice President
Entergy Operations, Inc.
Arkansas Nuclear One
1448 SR 333
Russellville, AR 72802-0967

SUBJECT: ERRATA FOR ARKANSAS NUCLEAR ONE – NRC AUGMENTED INSPECTION TEAM FOLLOW-UP REPORT 05000313/2013012 AND 05000368/2013012

Dear Mr. Browning:

Please remove pages A3-8 and A3-9 from the NRC Inspection Report 05000313/2013012 and 05000368/2013012 and replace them with the pages enclosed with this letter. The purpose of this change is to correct an administrative error in the detailed risk evaluation associated with Unit 2.

In accordance with Title 10 of the *Code of Federal Regulations* 2.390, "Public Inspections, Exemptions, Requests for Withholding," 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's Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Gregory E. Werner, Chief
Project Branch E
Division of Reactor Projects

Dockets No.: 50-313; 50-368
Licenses No.: DRP-51; NPF-6

Enclosure: Inspection Report 05000313/2013012; 05000368/2013012 Pages A3-8 and A3-9

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SUNSI Rev Compl.	☒ Yes ☐ No	ADAMS	☒ Yes ☐ No	Reviewer Initials	JLD
Publicly Avail	☒ Yes ☐ No	Sensitive	☐ Yes ☒ No	Sens. Type Initials	JLD

BC:DRS/EB2	BC:DRP/E			
JDixon	GWerner			
/RA/	/RA/			
4/8/14	4/9/14			

OFFICIAL RECORD COPY T=Telephone E=E-mail F=Fax the failure of once-through cooling. The evaluation of consequential loss of offsite power provided a dominant accident sequence involving a transient with consequential loss of offsite power, the loss of all feedwater to the steam generators and failure of once-through cooling.

Table 2
Core Damage Sequences

Sequence	Description	Point Estimate	% of Total	Cut Set Count
MFW-14	IEMFW-FW-OTC	2.69E-5	95.6	6,036
LOOP-19	IELOOP-EFW-OTC	3.79E-7	1.3	1,733
LOOP-20-09-10	IELOOP-SBO(EPS)-RSUB-OPR08H-DGR08H-EFWMAN-SGDEPLT	2.74E-7	1.0	527
MFW-15-10	IEMFW-RPS-FWATWS	1.25E-7	0.4	157
MFW-13	IEMFW-FW-SSRC-HPR	8.98E-8	0.3	1,679
LOOP-20-30	IELOOP-SBO-EFW-OPR08H-DGR08H	8.00E-8	0.3	959
MFW-02-09-04	IEMFW-LOSC-RCPT-HPI	6.14E-8	0.2	814
MFW-15-11	IEMFW-RPS-RCSPRESSURE	3.99E-8	0.1	18
MFW-15-09	IEMFW-RPS-BORATION	3.79E-8	0.1	16
MFW-12	IEMFW-FW-SSCR-CSR	2.63E-8	0.1	560
Others	All Additional Sequences Combined	1.33E-7	0.5	3,886
Total CCDP	All Sequences	2.81e-5	100.0	16,385

Abbreviations

BORATION	Failure of Emergency Boration
CBO	Controlled Bleedoff Isolated
CSR	Containment Spray Recirculation
DGR08H	Nonrecovery of Diesel Generator in 8 Hours
EFW	Emergency Feedwater
EFWMAN	Manual Control of Emergency Feedwater
EPS	Emergency Power System
FW	Feedwater System (MFW, EFW, and auxiliary feedwater)
FWATWS	Feedwater System under ATWS Conditions
HPI	High Pressure Injection
HPR	High Pressure Recirculation
IELOOP	Initiating Event: Loss of Offsite Power
IEMFW	Initiating Event: Loss of Main Feedwater
LOSC	Loss of RCP Seal Cooling
OPR08H	Nonrecovery of Offsite Power in 8 Hours
OTC	Once-Through Cooling
RCPT	Reactor Coolant Pumps Tripped
RCSPRESS	RCS Pressure Limited
RSUB	Reactor Coolant Subcooling Maintained
RPS	Reactor Protection System
SBO	Station Blackout
SGDEPLT	Late Depressurization of Steam Generators
SSCR	Secondary Cooling Recovered

The dominant accident sequence cutsets involved a loss of main feedwater, loss of auxiliary feedwater, loss of emergency feedwater, and the failure of once-through cooling. The top ten sequence cutsets are provided in Table 2 of the detailed risk evaluation.

The results are dominated by one core damage sequence. The largest contributor is Sequence 14 from the loss of main feedwater tree. The sequence comprises a failure of all feedwater to the steam generators, including main feedwater, auxiliary feedwater, and emergency feedwater, with a loss of once-through cooling. The remainder of the sequences are dominated by failure of the emergency diesel generators without recovery of ac power.

(6) Sensitivity Analysis

The SRA performed a variety of uncertainty and sensitivity analyses on the internal events model as shown below. The results confirm the recommended Yellow finding.

Sensitivity Analysis 1 – Transient without Loss of Main Feedwater.

The SRA ran the model using a transient as the initiator. The change in core damage frequency was 1.10×10^{-5} (Yellow).

Sensitivity Analysis 2 – No consequential loss of offsite power.

The SRA ran the model without including the additional runs to calculate the change in risk from a postulated consequential loss of offsite power. The change in core damage frequency was 2.74×10^{-5} (Yellow).

Sensitivity Analysis 3 – Potential Recovery of Bus 2A2

The SRA ran the model with the failure of Bus 2A2 probability set to 6.79×10^{-1}. This value, calculated using SPAR-H methodology, represented the probability that operators would fail to recover the bus prior to core damage, given the adverse and unknown conditions of site electrical supply. The change in core damage frequency was 1.97×10^{-5} (Yellow).

(7) Contributions from External Events (Fire, Flooding, and Seismic)

Manual Chapter 0609, Appendix A, Section 6.0 requires, "when the internal events detailed risk evaluation results are greater than or equal to 1.0E-7, the finding should be evaluated for external event risk contribution." The analyst noted that this detailed risk assessment evaluates an actual event in which no external events occurred. Additionally, the period of time that the events impacted plant equipment was small enough that the probability of an external initiator occurring during this time would be negligible. Therefore, the analyst assumed that the risk from external events, given the subject performance deficiency was essentially zero.