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

August 2, 2011

Mr. Mark Bezilla
Site Vice President
FirstEnergy Nuclear Operating Company
Perry Nuclear Power Plant
P. O. Box 97, 10 Center Road, A-PY-A290
Perry, OH 44081-0097

SUBJECT: PERRY NUCLEAR PLANT – NRC INSPECTION REPORT 05000440/2011009 AND 07200069/2011001 (DNMS)

Dear Mr. Bezilla:

On July 8, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed its inspection of the dry cask storage pad design at the Perry Nuclear Power Plant. The enclosed report presents the inspection findings which were discussed on July 8, 2011, with you and other members of your staff.

The inspection examined the dry fuel storage pad design, and connecting pathway, as it relates to the safe storage of dry fuel storage casks and compliance with the Commission's rules, regulations, and the conditions of your license. Specific areas examined during the inspection are identified in the enclosed report. Within these areas, the inspection consisted of selected examinations of procedures and representative records, and interviews with personnel.

The inspection was conducted per NRC Inspection Manual 2690, "Inspection Program for Dry Storage of Spent Reactor Fuel at Independent Spent Fuel Storage Installations and Guidance for Title 10 of the Code of Federal Regulations (CFR) Part 71 Transportation Packages," and used portions of Inspection Procedure (IP) 60853 and IP 60856.

Based on the results of this inspection, the inspectors indentified two findings of very low safety significance. Both of these findings involved violations of NRC requirements. However, because these violations were of very low safety significance, and were entered into your corrective action program, the NRC is treating these issues as non-cited violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

If you contest the subject or severity of the NCVs, 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 Perry Nuclear Power Plant. In addition, if you disagree with the cross-cutting aspect assigned to 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 Perry Nuclear Power Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and the enclosed report will be available electronically for public inspection in the NRC Public Document Room or from the NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

Sincerely,

/RA/

Christine A. Lipa, Chief
Materials Control, ISFSI, and
Decommissioning Branch
Division of Nuclear Materials Safety

Docket Nos: 72-069 and 50-440
License No: NPF-58

Enclosure:
NRC Inspection Reports 07200069/2011001(DNMS) and
05000440/2011009

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos.	72-069; 50-440
License No.	NPF-58
Report Nos.	07200069/2011001(DNMS) and 05000440/2011009
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Perry Nuclear Power Plant, Unit 1
Location:	Perry, OH
Inspection Dates:	Onsite: October 14 – 16, 2009; November 10 – 12, 2009; April 5 – 7, 2010; December 1 – 3, 2010; July 5 – 8, 2011. In-Office completed July 8, 2011
Inspectors:	John Bozga, Reactor Inspector Rhex Edwards, Reactor Inspector
Approved by:	Christine A. Lipa, Chief Materials Control, ISFSI, and Decommissioning Branch Division of Nuclear Materials Safety

Enclosure

SUMMARY OF FINDINGS

IR 07200069/2011001(DNMS) and 05000440/2011009; 10/14/2009 – 07/08/2011; Review of 10 CFR 72.212(b) evaluations and ISFSI Storage Pad Design

The purpose of the inspection was to evaluate the design of a new Independent Spent Fuel Storage Installation (ISFSI) storage pad and haul path, at the Perry Nuclear Plant, to ensure compliance with regulations and design specifications. This inspection began on October 14, 2009 and ended July 8, 2011. One Green finding and one Severity Level IV violation were identified by the inspectors. The U.S. Nuclear Regulatory Commission (NRC) is treating both issues as non-cited violations (NCVs) 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). The cross-cutting aspects were determined using IMC 0310, "Components Within the Cross-Cutting Areas." 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

- Severity Level IV. A finding of very low safety-significance and associated Severity Level IV non-cited violation (NCV) of 10 CFR 72.212 (b)(2)(i), "Conditions of a General License Issued under 72.210," was identified by the inspectors for the failure of the licensee to incorporate American Concrete Institute (ACI) code requirements and American Society of Civil Engineer's (ASCE) standards into the design bases of the ISFSI pad and for not evaluating the potential impact a high mast light, not capable of withstanding site specific tornado wind loads, would have on the storage casks located on the ISFSI pad. This has been entered into the licensee's corrective action program as Condition Report (CR)10-86678, CR11- 88793, and CR10-86590.

  The inspectors determined that the issue was of more than minor significance using Example 3k of Appendix E, "Examples of Minor Issues" of IMC 0612, "Power Reactor Inspection Reports." Specifically, the licensee made assumptions in the written analyses that, if left uncorrected, lead to reasonable doubt as to the structural integrity of the ISFSI pad during a postulated seismic event and the integrity of the storage overpack (HI-STORM) following the falling of a high mast light due to potential site specific tornado winds. The inspectors determined that the issue could be evaluated using example 6.5.d.1 of the NRC Enforcement Policy as a Severity Level IV violation in that the licensee failed to meet requirements that have more than minor significance. (Section 4OA5)

  Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance and an associated non-cited violation of Title 10 of the Code of Federal Regulations (CFR) Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for failure to perform an adequate evaluation for Emergency Service Water (ESW) system piping. Specifically, the inspectors identified that the licensee had not evaluated all design and licensing basis loads and load combinations in accordance with Seismic Category I and American Society of Mechanical Engineers (ASME) code requirements. The licensee documented the corrective actions in CR10-86678 and CR11-88800.

The inspectors determined that the performance deficiency affected the Mitigating Systems Cornerstone. The inspectors compared this performance deficiency to the minor questions of IMC 0612, Appendix B, "Issue Screening," dated December 24, 2009, and the inspectors determined that this finding was more than minor because, if left uncorrected, the failure to perform an adequate evaluation of the ESW system piping would have the potential to become a more significant safety concern. Absent NRC intervention, the licensee would not have performed the evaluation of the Vertical Cask Transporter (VCT) load in combination with seismic load as well as other design basis loads which would have placed the piping in a potential overstress condition leading to a permanent deformation of the piping where the system would not be able to perform its safety function and it would become a more significant safety concern. Specifically, compliance with Seismic Category I and ASME code requirements was to ensure structural integrity of the ESW piping during a design basis event. 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," Table 4a for the Mitigating Systems Cornerstone. The inspectors answered "yes" to the question of is the finding a design qualification deficiency confirmed not to result in loss of operability or functionality in the Mitigating Systems column based on the licensee revising design calculations and initiated modifications where necessary to demonstrate compliance and concluded that the finding was of very low safety-significance (Green). The inspectors identified a Human Performance, Work Practices (H.4.c) cross-cutting aspect associated with this finding. The licensee did not ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety was supported. Specifically, the licensee failed to have effective oversight of design calculation and documentation for demonstrating ASME code compliance of the ESW system piping. [H.4(c)] (Section 4OA5)

B. <u>Licensee-Identified Violations</u>

No violations of significance were identified.

REPORT DETAILS

4.0 OTHER ACTIVITIES (OA)

4OA5 Other Activities

Review of 10 CFR 72.212(b) Evaluations: Review of ISFSI Storage Pad Design, Dry Cask Transfer Route, and Design Basis Events (IP 60856)

a. Inspection Scope

The objective of this inspection was to determine whether the requirements, as they relate to the ISFSI pad and haul path, specified in section 72.212 of 10 CFR, “Conditions of General License issued under 72.210,” have been met by the Perry Nuclear Power Plant. In general, the inspection assessed the licensee’s geologic, seismic, tornado, and flooding evaluations to verify the licensee’s compliance with the Certificate of Compliance, 10 CFR 72 requirements, and industry standards. In-office reviews, walk-downs, and discussions with site personnel were conducted during the inspection. The specific areas inspected include:

(1) Soil Analysis

The inspectors evaluated whether the reactor site soil structure differed from the soil structure under the ISFSI storage pad through in-office document reviews. This evaluation included reviews of test borings and Cone Penetration Tests performed for the ISFSI storage pad areas. Soil compaction of the soil underneath the ISFSI pad was also reviewed.

(2) Seismic Analysis

The inspectors reviewed the licensee’s seismic analysis evaluation to determine if the site’s safe shutdown earthquake (SSE) accelerations were correctly considered at the ISFSI. This analysis was compared to the design basis specified in the Holtec HI-STORM 100S Final Safety Analysis Report (FSAR), Revision 7. Included in this review, the inspectors evaluated the site’s conclusion regarding potential sliding and tipping of a storage cask during a seismic event. Additionally, the effects of pad settlement were reviewed during static and seismic events, as well as, the effects of partial and full loads on the soils bearing capacity. The site’s soil-structure interaction analysis was reviewed to determine that the ISFSI storage pad will adequately support both static and dynamic loads, as required by 10 CFR 72.212 (b)(2)(ii) and 72.212(b)(3).

(3) Liquefaction Analysis

The potential for soil liquefaction was evaluated and reviewed by the inspectors using the empirical method described in Regulatory Guide (RG) 1.198. The inspectors reviewed the site’s seismic ground motion accelerations and how they were utilized in the soil liquefaction analysis.

(4) Flooding

A review of the site's hydrological data was performed including the effects of flooding the ISFSI site from Lake Erie and the effects on pertinent structures that would impact the ability to safely conduct ISFSI operations. The inspectors also reviewed the drainage pathway and the catch basins that were modified following construction of the ISFSI pad.

(5) Dry Cask Transfer Route

The dry cask transfer route, or haul path, is the pathway where a loaded HI-STORM is transported from the PNPP Fuel Handling Building (FHB) to the ISFSI pad. The inspectors reviewed the site's evaluation of the maximum load traversing the haul path. Additionally, the inspectors reviewed the buried utilities that are buried beneath the haul path and the site's evaluation of the maximum load traversing over them.

(6) Tornado Analysis

The inspectors reviewed the site's evaluations of the HI-STORM following a potential site specific tornado hazard to determine whether it was bounded by the Holtec HI-STORM FSAR, as required by 10 CFR 72.212(b)(3).

b. Findings

(1) Cask Storage Pad Evaluations Did Not Meet 10 CFR 72.212(b)(2) Requirements

Introduction: A finding of very low safety-significance and associated Severity Level IV Non-Cited Violation (NCV) of 10 CFR 72.212 (b)(2)(i),"Conditions of a General License Issued under 72.210," was identified by the inspectors for the failure of the licensee to incorporate industry code requirements and regulatory guidance into the design bases of the ISFSI pad and determinations that effects of potential site tornado hazards are enveloped by the cask design bases.

Description: Title 10 CFR 72.212(b)(2)(i)(B) requires that written evaluations be performed to establish that the cask storage pads and areas have been designed to adequately support the static and dynamic loads of the stored casks, considering potential amplification of earthquakes through soil-structure interaction, soil liquefaction potential, or other soil instability due to vibratory ground motion.

Specifically, the inspectors identified two examples where the licensee's evaluations failed to demonstrate that the ISFSI pad was designed to adequately support the static and dynamic loads of the stored casks, considering potential amplification of earthquakes through soil-structure interaction, soil liquefaction potential, or other soil instability due to vibratory ground motion. In addition, the inspectors identified one example where the damage to a HI-STORM, located on the ISFSI pad, was not considered due to a potential site specific tornado wind event.

(a) On July 11, 2009, the licensee completed calculation number G58-S-G-005, "Static Settlements at Independent Spent Fuel Storage Installation (ISFSI) Site," Revision 0. ACI 349-85, Section 9.2.2, requires where the structural effects of differential settlement may be significant, they shall be considered in specified load combinations. Differential settlement was evaluated in calculation number G58-S-G-005; however, the effects were not evaluated for compliance with ACI 349-85 and therefore were not included in the design of the ISFSI pad. Calculation number G58-S-SC-006, “Design of the ISFSI Pad for the HI-Storm Vertical Storage Casks,” Revision 0, specifies that the strength requirements from ACI349-85 be utilized in the design of the ISFSI pad. CR10-86678, “Out of Plane Flexibility and Differential Settlement not in some SFDS [Spent Fuel Dry Storage]Pad Calcs,” dated December 6, 2010, was generated to address this issue.

(b) On November 10, 2009, the licensee completed calculation number G58-S-SC-005, “Seismic Analysis of HOLTEC HI-STORM Storage Modules on Perry NPP ISFSI Base Mat,” Revision 0. The calculation contains a Soil Structure Interaction (SSI) analysis that assumes the storage casks remain in contact with the storage pad during a seismic event and that the pad behaves as a rigid body. The assumption that the pad behaves as a rigid body was justified to be in accordance with Section 3.3.1.6 of ASCE standard 4-98, “Seismic Analysis of Safety-Related Nuclear Structures and Commentary.” The ASCE standard 4-98 commentary states that “For typical nuclear power plant structures, the effect of mat flexibility for mat foundations… need not be considered in SSI analysis. Although foundations and walls may appear to be flexible when taken by themselves, an effective stiffness of the foundation must be evaluated to adequately assess its flexibility. The effective stiffness is a function of the foundation itself and the stiffening effect of structural elements tied to the foundation. The latter item contributes significant stiffening effects in typical nuclear power plant containment and shear wall structures.” Since there are no structural elements tied to the pad foundation, the pad must be considered flexible and its effects considered. The influence of pad out-of-plane flexibility on seismic response of the casks and the seismic demand on the ISFSI pad was not addressed in calculation number G58-S-SC-005, Revision 0, and therefore was not included in the design of the ISFSI pad. Condition Report (CR) 10-86678, “Out of Plane Flexibility and Differential Settlement not in some SFDS Pad Calcs,” dated December 6, 2010, was generated to address this issue.

(c) On July 1, 2009, the site completed calculation number G58-S-SC-008, “Design of Lighting Mast No. 7 Foundation,” Revision 0, for the foundation design of High Mast Light (HML) No. 7. The HML No. 7 was not designed to withstand a site specific tornado wind event and was not evaluated for its failure during a potential tornado wind event and subsequent impact on a HI-STORM located on the ISFSI pad. Therefore, it was not shown that the reactor site parameters bounded the cask design basis as required to be evaluated per 72.212(b)(3). To address this issue, CR11-88793, “Evaluation of HML #7 Impact on HI-STORMS for NRC Questions 39 and 242,” dated January 26, 2011, and CR10-86590, “HI-MAST Light #7 Not Evaluated as Tornado Missile for HI-STORMS,” dated December 3, 2010, were generated.

As a result of the inspector’s concerns, documented in these examples, the licensee was performing revisions to their evaluations.

Analysis: The inspectors determined that not considering the effects of differential settlement, pad flexibility, and hazards created from potential tornado winds was a performance deficiency that warranted a significance evaluation. Consistent with the guidance in Section 2.2 of the NRC Enforcement Policy, ISFSIs are not subject to the Significance Determination Process and, thus, traditional enforcement will be used for this issue. The inspectors determined that the violation was of more than minor significance using Example 3k of Appendix E, "Examples of Minor Issues" of IMC 0612, "Power Reactor Inspection Reports." Consistent with the guidance in Section 2.6.D of the NRC Enforcement Manual, if a violation does not fit an example in the Enforcement Policy Violation Examples, it should be assigned a severity level: (1) Commensurate with its safety significance; and (2) informed by similar violations addressed in the Violation Examples. The inspectors determined that the violation could be evaluated, using Section 6.5.d.1 of the NRC Enforcement Policy, as a Severity Level IV Violation because the licensee failed to meet a regulatory requirement that has more than a minor safety significance.

<u>Enforcement</u>: Title 10 CFR 72.212 (b)(2)(i)(B) requires, in part, that the licensee perform written evaluations prior to use, that establish that cask storage pads and areas have been designed to adequately support the static and dynamic loads of the stored casks, considering potential amplification of earthquakes through soil structure interaction, and soil liquefaction potential or other soil instability due to vibratory ground motion. Furthermore, 72.212 (b)(3) requires, in part, that the licensee determine whether or not the reactor site parameters, including analyses of earthquake intensity and tornado missiles, are enveloped by the cask design bases. The results of this review must be documented in the evaluation performed under 10 CFR 72.212(b)(2)(i)(b).

Contrary to the above, on July 11, 2009, and November 10, 2009, the licensee's evaluations failed to demonstrate that the ISFSI pad was designed to adequately support the static and dynamic loads of the stored casks, considering potential amplification of earthquakes through soil-structure interaction, soil liquefaction potential, or other soil instability due to vibratory ground motion. In addition, on July 1, 2009 an evaluation of the HML falling, due to potential tornado winds, on a HI-STORM did not determine whether reactor site parameters were enveloped by the cask design basis. This is a violation of 10 CFR 72.212 (b)(2)(i)(B), "Conditions of a General License Issued under 72.210." There are no actual safety consequences since dry fuel storage canisters have not been placed on the ISFSI pad. The licensee was performing revisions to their evaluations and has entered the issues into their corrective action program (CR10-86678, CR11-88793, and CR10-86590). Because this matter was of very low safety-significance (Severity Level IV), and has been entered into the licensee's corrective action program, this violation is being treated as an NCV consistent with the Enforcement Policy (NCV 07200069/2011001-01).

(2) <u>Emergency Service Water System Piping did not meet ASME Code Requirements</u>

<u>Introduction</u>: A finding of very low safety significance (green) and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for failure to perform adequate evaluations for the ESW system piping used to support the spent fuel cask transport. Specifically, the inspectors identified where the licensee failed to perform adequate evaluations of the ESW piping in accordance with Seismic Category I and ASME code requirements as defined in the Perry Updated Final Safety Analysis Report (UFSAR) Section 9.2.1.

<u>Description</u>: The ESW system was required to be Seismic Category I per UFSAR Section 9.2.1. The code used for Seismic Category I compliance for the ESW system

piping was the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plants components as described in UFSAR Section 9.2.1.

The ESW System was designed to provide a reliable source of water to safety-related components required for certain modes of normal operation, as well as for accident conditions and loss of normal auxiliary power. The function of the ESW System was designed to provide a reliable source of service water to safety related components required for certain modes of normal reactor operation, as well as under accident conditions and loss of normal auxiliary power. Specific components supplied with service water are the residual heat removal (RHR) heat exchangers, the high pressure core spray (HPCS), standby diesel generators, the emergency closed cooling heat exchangers, and the HPCS room cooler.

The ASME Boiler and Pressure Vessel Code Section III, 1974 edition including Addenda through winter of 1975, Subsection ND was the design and licensing basis code used for the analysis of the ESW system piping. ASME Section III, Subsection ND-3112, "Design Conditions," states "The components shall be designed in accordance with the owner's Design Specification (NA-3250)." The ASME Section III, Subsection NA-3250, "Provisions of Design Specification," states "It is the responsibility of the Owner to provide, or cause to be provided, Design Specifications for components, appurtenances, and component supports. The Owner, either directly or through his designee, shall be responsible for the proper correlation of all Design Specifications. Separate Design Specifications are not required for parts, piping subassemblies, appurtenances, or component supports when they are included in the Design Specification for the component."

Specification Number DSP-P45, "Emergency Service Water (ESW) System ASME Design Specification," Revision 5, contained information regarding the ESW System (P45) piping and pipe support components, meeting the requirements of the 1974 ASME B & PV Code, Section III, Division 1, with addenda up to and including the winter 1975 issue.

Specification Number DSP-P45, Revision 5, defined the design loadings in Section 3:02.2 for the ESW System. Section 3:02.2 defines other mechanical loads as unbalanced forces, applied external loads, earth loading, vehicle loading, etc. The earthquake loading defined in Section 3:02.2 was based on two earthquake models: the Operating Basis Earthquake (OBE) and SSE. The loading combinations in Table 3B, "ASME Code Analysis Load Combination and Stress Limit Design Criteria for Class 2 and 3 Piping Systems Perry Nuclear Power Plant Emergency Service Water System" and Table 3C, "Summary of Load Combination and Stress Limit Design Criteria for Class 1, 2 and 3 Piping Systems Perry Nuclear Power Plant Emergency Service Water System," required the evaluation of combined load effects due to other mechanical loads such as vehicle loading in combination with earthquake loading due to either an OBE or a SSE.

Calculation Number G58-S-SY-002, "Evaluation of Buried Items at the ISFSI Site with Vertical Cask Transporter," Revision 1, evaluated the emergency service water system piping for the VCT load and dead load due to soil weight and surcharge. The calculation stated "Total load acting on the pipe is the sum of the dead load (DL) due to soil weight and the surcharge or live load (LL) due to the VCT. Based on the temporary nature of the load, the VCT surcharge load is not combined with the seismic load. Therefore, only the DL + LL case is evaluated." The licensee's use of probability did not meet Seismic Category I and ASME code requirements as defined in Specification number DSP-P45. The inspectors identified that the licensee did not consider seismic load concurrent with VCT load and the licensee documented these deficiencies in CR10-86678, "Out of Plane Flexibility and Differential Settlement Not in Some SFDS Pad Calcs," dated December 6, 2010 and CR11-88800, "NRC Dry Fuel Questions 262 202 140 141 and 27 not separately noted in CR10-86678," dated January 26, 2011.

Analysis: The inspectors determined that the licensee's failure to perform an adequate evaluation to demonstrate Seismic Category I and ASME code compliance for the ESW system piping was contrary to the design control measures per 10 CFR Part 50 Appendix B requirements and was a performance deficiency.

The inspectors determined that the performance deficiency affected the Mitigating Systems Cornerstone. As a result, the inspector compared this performance deficiency to the minor questions of IMC 0612, Appendix B, "Issue Screening," dated December 24, 2009, and determined that this finding was more than minor because, if left uncorrected, the failure to perform an adequate evaluation of the ESW system piping would have the potential to become a more significant safety concern. Absent NRC intervention, the license would not have performed an evaluation of the VCT load concurrent with seismic load as well as other design basis loads. This would have placed the piping in a potential overstress condition leading to permanent deformation of the piping where the system would not be able to perform its safety function.

Specifically, compliance with Seismic Category I and ASME code requirements was to ensure structural integrity of the emergency service system piping during a design basis event.

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," Table 4a for the Mitigating Systems Cornerstone. The inspectors answered "yes" to the question of is the finding a design qualification deficiency confirmed not to result in the loss of operability or functionality in the Mitigating Systems column based on the licensee revising design calculations and initiating modifications where necessary to demonstrate compliance and concluded that the finding was of very low safety-significance (Green).

The inspectors identified a Human Performance, Work Practices (H.4.c) cross-cutting aspect associated with this finding. The licensee did not ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety was supported. Specifically, the licensee failed to have effective oversight of design calculation and documentation for demonstrating ASME code compliance of the ESW system piping. (H.4(c)).

Enforcement: Title 10 CFR 50, Appendix B, Criterion III, "Design Control" states, in part, that the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Contrary to the above, on December 6, 2010, in Calculation No. G58-S-SY-002, Revision 1, the inspectors determined that the licensee's design control measures failed to verify adequacy of the emergency service water system piping. The licensee did not consider the combined effects of seismic load and VCT load and did not meet ASME code requirements.

Because this violation was of very low safety significance (Green) and it was entered into the licensee's corrective action program as CR10-86678 and CR11-88800, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000440/2011009-01)

4OA6 Management Meetings

Exit Meeting Summary

On July 8, 2011, the inspectors presented the inspection results to Mr. M. Bezilla 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.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

M. Bezilla, Vice President Nuclear
N. Bonner, Dry Cask Storage Project
C. Elberfeld, Supervisor Nuclear Compliance
J. Fox, Dry Cask Storage Project Manager
J. Grabnar, Site Engineering Director
D. Haviland, Dry Cask Storage Project
B. Spiesman, Fleet Licensing
P. Wilson, Dry Cask Storage Project
L. Zerr, Nuclear Compliance

INSPECTION PROCEDURES USED

IP 60853 Construction of an Independent Spent Fuel Storage Installation

IP 60856 Review of 10 CFR 72.212 (b) Evaluations, Appendix A, Review of ISFSI Storage Pad Design

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened:

07200069/2011001-01	NCV	Cask Storage Pad Evaluations did not meet 10 CFR 72.212(b)(2) Requirements (Section 4OA5)
05000440/2011009-01	NCV	Emergency Service Water System Piping did not meet ASME Code Requirements (Section 4OA5)

Closed:

07200069/2011001-01	NCV	Cask Storage Pad Evaluations did not meet 10 CFR 72.212(b)(2) Requirements (Section 4OA5)
05000440/2011009-01	NCV	Emergency Service Water System Piping did not meet ASME Code Requirements (Section 4OA5)

Discussed: None

LIST OF DOCUMENTS REVIEWED

The following is a partial 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.

4OA5 Other Activities

- G58-S-SY-001; “Drainage Calculation for Proposed Independent Spent Fuel Storage Installation;” Revision 0
- G58-S-SY-002; “Evaluation of Buried Items at the ISFSI Site with Vertical Cask Transporter;” Revision 1
- G58-S-SY-003; “Road Design of ISFSI Operating Area;” Revision 0
- G58-S-R-Y-005; “Evaluation of the Perry ISFSI Location and Arrangement Options;” Revision 0
- G58-S-SC-004; “Design of the ISFSI Haul Route Turning Slab and Bridging Slab;” Revision 0
- G58-S-SC-005; “Seismic Analysis of HOLTEC HI-STORM Storage Modules on Perry NPP ISFSI Base Mat;” Revision 0
- G58-S-SC-006; “Design of the ISFSI Pad for HI-STORM Vertical Storage Casks;” Revision 0
- G58-S-SC-008; “Design of Lighting Mast No. 7 Foundation;” Revision 0
- G58-S-SC-012; “Seismic Input Verification for the Perry NPP ISFSI Analysis;” Revision 0
- G58-S-SC-013; “Generation of Seismic Response Spectra for the Perry NPP ISFSI Pad;” Revision 0
- G58-S-G-001; “Development of Report-Quality Boring Logs & Subsurface Profiles;” Revision 0
- G58-S-G-002; “Bases of Geotechnical Parameters Recommended for Design;” Revision 0
- G58-S-G-003; “Liquefaction Analysis and Estimation of Post-Earthquake Settlements at Independent Spent Fuel Storage Installation (ISFSI) Site;” Revision 0
- G58-S-G-004; “ISFSI Pad Overturning, Sliding, and Bearing Capacity Analyses;” Revision 0
- G58-S-G-005; “Static Settlements at Independent Spent Fuel Storage Installation (ISFSI) Site;” Revision 0
- G58-S-R-G-009; “Geophysical Investigation;” Revision 0
- G58-S-R-G-010; “Geotechnical Design Report for the Proposed Independent Spent Fuel Storage Installation (ISFSI);” Revision 0
- G58-H-HI-2094384; Missile Penetration Analysis for Perry HI-STORM;” Revision 0
- CR10-86590; “HI-MAST Light #7 Not Evaluated as Tornado Missile for HI-STORMS;” dated December 3, 2010
- Specification No. DSP-P45; “Emergency Service Water (ESW) System ASME Design Specification;” Revision 5
- CR10-86673; “East Yard Storm Drain Calculation Contains an Error;” dated December 6, 2010
- CR10-86678; “Out of Plane Flexibility and Differential Settlement not in some SFDS Pad Calcs;” dated December 6, 2010
- CR11-88793; “Evaluation of HML #7 Impact on HI-STORMS for NRC Questions 39 and 242;” dated January 26, 2011
- CR11-88800; “NRC Dry Fuel Questions 262 202 140 141 AND 27 not separately noted in CR10-86678;” dated January 26, 2011

- CR11-90858; “NRC Dry Fuel Inspection Question 276 Potential Deviation of ASME Code Reqmnts;” dated March 11, 2011
- CR11-88915; “NRC SFDS Questions and Inspection Closure Verification Concern;” dated January 28, 2011

LIST OF ACRONYMS USED

ACI	American Concrete Institute
ADAMS	Agencywide Document Access Management System
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CR	Condition Report
DL	Dead Load
ESW	Emergency Service Water
FHB	Fuel Handling Building
FSAR	Final Safety Analysis Report
HML	High Mast Light
HPCS	High Pressure Core Spray
IP	Inspection Procedure
ISFSI	Independent Spent Fuel Storage Installation
LL	Live Load
NCV	non-cited violation
NRC	U.S. Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
RG	Regulatory Guide
RHR	Residual Heat Removal
SDP	Significance Determination Process
SFDS	Spent Fuel Dry Storage
SSE	Safe Shutdown Earthquake
SSI	Soil Structure Interaction
UFSAR	Updated Safety Analysis Report
VCT	Vertical Cask Transporter

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 Perry Nuclear Power Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and the enclosed report will be available electronically for public inspection in the NRC Public Document Room or from the NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

Sincerely,

Christine A. Lipa, Chief
Materials Control, ISFSI, and
Decommissioning Branch
Division of Nuclear Materials Safety

Docket Nos: 72-069 and 50-440
License No: NPF-58

Enclosure:
NRC Inspection Reports 07200069/2011001(DNMS) and 05000440/2011009

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